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The reactor core is a multi-region core. The fuel rods are cold-worked partially annealed Zircaloy tubes containing slightly enriched uranium dioxide fuel. All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.

The fuel assembly is a canless type with the basic assembly consisting of the Rod Cluster Control (RCC) guide thimbles fastened to the grids and the top and bottom nozzles. The fuel rods are supported at several points along their length by the spring-clip grids.

Full-length Rod Cluster Control Assemblies (RCCAs) are inserted into the guide thimbles of selected fuel assemblies. The absorber sections of the RCCA are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes. Gadolinia is used as an integral burnable poison in selected fuel assemblies. Discrete burnable poison rods may also be used. The absorber material in the discrete burnable poison rods is in the form of borosilicate glass sealed in stainless steel tubes.

The RCC drive mechanisms for full-length rods are of the magnetic latch type. The latches are controlled by three magnetic coils. They are so designed that, upon loss of power to the coils, the RCCA is released and falls by gravity to shut down the reactor.

3.1 DESIGN BASES

3.1.1 Performance Objectives

The construction permit was issued for an initial reactor power of 1650 MWt with an ultimate rating of 1721.4 MWt. The original license application was for the 1650 MWt power rating. The 1721.4 MWt power level was the basis, except where specifically noted, for all the safeguard evaluations in this report. In 2003, a measurement uncertainty recapture (MUR) power uprate was performed that increased the licensed rated power from 1650 MWt to 1673 MWt (Reference 1).

In 2004, using available plant design margin, a 6 percent stretch power uprate was performed that increased the licensed rated power from 1673 MWt to 1772 MWt (Reference 2). The safety analyses that support the 6 percent stretch power uprate are described in Chapter 14. The reactor core fuel loading is designed to yield an 18-month cycle burnup of approximately 17,500 MWD/MTU.

The RCCAs provide sufficient control rod worth to shut the reactor down by the shutdown reactivity required by Technical Specifications in the hot condition at any time during the fuel cycle with the most reactive RCCA stuck in the fully withdrawn position. Redundant equipment is provided to add soluble poison to the reactor coolant to ensure a similar shutdown capability when the reactor coolant is cooled to ambient temperatures.

Experimental measurements from critical experiments and operating reactors were used to validate the methods employed in the design. In the reload core design, nuclear parameters are calculated for every phase of operation of the cycle and, where applicable, are compared with design limits to show that an adequate margin of safety exists.

In the thermal-hydraulic design of the reload core, the maximum fuel and clad temperature during normal reactor operation and at thermal overpower conditions are conservatively evaluated to verify that operation is within existing safety limits.

3.1.2 Principal Design Criteria

3.1.2.1 Reactor Core Design

Criterion: The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated (GDC 6).

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and residual heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations.

The Reactor Control and Protection System is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a Departure from Nucleate Boiling Ratio (DNBR) equal to or greater than the applicable DNBR correlation limit.

The integrity of the fuel cladding is ensured by preventing excessive fuel swelling, excessive cladding overheating, and excessive cladding stress and strain. This is achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

1. DNBR equal to or greater than the applicable DNBR correlation limit.
2. Fuel center temperature below melting point of UO_2.
3. Internal gas pressure less than the fuel manufacturer's approved limit even at the end of life.
4. Clad stresses less than the Zircaloy yield strength.

5. Clad strain less than 1 percent.
6. Cumulative strain fatigue cycles less than 80 percent of design strain fatigue life.

The ability of fuel rods designed to these limits to withstand postulated normal and abnormal service conditions is shown by the safety analyses described in Chapter 14.

3.1.2.2 Suppression of Power Oscillations

Criterion: The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed (GDC 7).

The potential for possible spatial oscillations of power distribution for the core has been reviewed. In summary it is concluded that the only potential spatial instability is the xenon-induced axial instability which may be an oscillation with little or no inherent damping. Out-of-core instrumentation is provided to obtain necessary information concerning axial distributions. This instrumentation is adequate to enable the operator to monitor and control xenon-induced oscillations. In-core instrumentation is used to calibrate and verify the axial flux information provided by the out-of-core instrumentation. The analysis, detection and control of these oscillations is discussed in Reference 1 of Section 3.2.1.

The moderator temperature coefficient is maintained zero or negative at power levels ≥60 percent of full power to provide further control of power oscillations.

3.1.2.3 Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided (GDC 27).

Two independent reactivity control systems are provided, one involving rod cluster control (RCC) assemblies and the other involving the injection of a soluble poison.

3.1.2.4 Reactivity Hot Shutdown Capability

Criterion: The reactivity control systems provided shall be capable of making and holding the core sub-critical from any hot standby or hot operation condition (GDC 28).

The reactivity control systems provided are capable of making and holding the core sub-critical from any hot standby or hot operating condition, including those resulting from power changes. This includes the maximum excess reactivity expected for the core, which occurs for the cold clean condition at the beginning of life of a reload core.

The RCCAs are divided into two categories comprising control and shutdown rod groups. The control group, used in combination with soluble poison, provides reactivity control throughout the life of the core at power conditions. This group of RCCAs is used to compensate for short-term reactivity changes due to variations in reactor power requirements or coolant temperature. The soluble poison control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup and for load-follow.

Upon demand for the hot shutdown condition, insertion of both the control and shutdown groups of RCCAs will immediately make the reactor sub-critical from any hot standby or hot operating condition. Subsequent injection of soluble poison can be used to assure continuation of the hot shutdown condition under all circumstances.

3.1.2.5 Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core sub-critical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure sub-criticality with the most reactive control rod fully withdrawn (GDC 29).

The reactor core, together with the reactor control and protection system, is designed so that the minimum allowable DNBR is greater than the applicable DNBR correlation limit and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups are provided to supplement the control group of RCCAs to make the reactor at least 1 percent sub-critical ($k_{eff} = 0.99$) following a trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCCA remains in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core sub-critical for the most severe anticipated cooldown transient associated with a single active failure; e.g., accidental opening of a steam bypass, or safety valve stuck fully open.

The criteria of GDC-28 and -29 are met fast enough to prevent exceeding acceptable fuel damage limits, even with the most reactive RCCA fully withdrawn.

3.1.2.6 Reactivity Holddown Capability

Criterion: The reactivity control system provided shall be capable of making the core sub-critical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public (GDC 30).

The reactivity control systems, provided are capable of making and holding the core sub-critical under accident conditions, in a timely fashion with appropriate margins for contingencies. Normal reactivity shutdown capability is provided within 2 seconds following a trip signal by the insertion of the RCCAs with soluble poison (boric acid) injection used to compensate for the long-term xenon decay transient and for plant cooldown. Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and Refueling Water Storage Tank (RWST) is ready for injection always exceeds that required for the normal cold shutdown condition. This quantity also exceeds that required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

Boric acid is pumped from the boric acid tanks by two boric acid pumps to the suction of three charging pumps, which inject boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel generator power on loss of off-site power. Boric acid can be injected by one pump at a rate, which places the reactor in the hot shutdown condition with no RCCAs inserted and also compensates for xenon decay. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

Based on the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of the RCCAs which normally serve this function in the short-term situation. Shutdown for long-term and reduced-temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at a lower concentration can be supplied from the refueling water storage tank (RWST). This solution can be transferred directly by the charging pumps. The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown. For added flexibility, the safety injection pumps can also be supplied with boric acid solution from the RWST, which has a minimum required boron concentration.

3.1.2.7 Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits (GDC 31).

The reactor protection systems are capable of protecting against any single anticipated malfunction of the reactivity control system; by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with RCCAs is completely independent of the normal RCCA control functions since the trip breakers completely interrupt the power to the latch-type rod mechanisms regardless of existing control signals. Details of the effects of continuous withdrawal of an RCCA and continuous deboration are described in Section 14.1 and Section 9.2, respectively.

3.1.2.8 Maximum Reactivity Worth of RCCAs

Criterion: Limits, which include reasonable margin shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot:

1. rupture the reactor coolant pressure boundary, or
2. disrupt the core, its support structures or other vessel internals sufficiently to lose capability of cooling the core (GDC 32).

Limits, which include considerable margin are placed on the maximum reactivity worth of RCCAs or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large reactivity change cannot:

1. rupture the reactor coolant pressure boundary or
2. disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The reactor control system employs RCCAs, which are divided into control and shutdown groups. During operation, the shutdown banks are positioned fully withdrawn from the core to provide negative reactivity for shutdown margin. The controlling groups are used to control load and reactor coolant temperature. The full-length RCC drive mechanisms are wired into pre-selected groups and are therefore prevented from being withdrawn in other than their respective groups. The RCC drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable-speed travel.

The maximum positive reactivity insertion rate assumed in the detailed plant analysis (Section 14.1) is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed. The resultant reactivity insertion rates are well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 72 steps per minute (~45 inches per minute).

3.1.3 Safety Limits

The reactor is capable of meeting the performance objectives throughout core life under both steady state and transient conditions without violating the integrity of the fuel cladding. Thus, the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation (Technical Specifications Section 3) specify the minimum functional capability or performance levels necessary to assure safe operation of the facility.

Design parameters, which are established by safety limits (Technical Specifications Section 2), are described below for the nuclear, reactivity control, thermal and hydraulic, and mechanical aspects of the design.

3.1.3.1 Nuclear Limits

At power, the nuclear heat flux hot channel factor, ${F^N}_q$, and the nuclear enthalpy rise hot channel factor, ${F^N}_{\Delta H}$, shall not exceed their respective limits as specified in the Core Operating Limits Report.

The nuclear axial peaking factor ${F^N}_z$ and the nuclear enthalpy rise hot channel factor ${F^N}_{\Delta H}$ are limited in their combined relationship so as not to exceed ${F^N}_q$ and minimum departure from nucleate boiling ratio (DNBR) limits.

The limiting nuclear hot channel factors are higher than those calculated at full power for the range from all RCCAs fully withdrawn to maximum allowable RCCA insertion. RCCA insertion limits as a function of power are delineated in the Core Operating Limits Report to ensure that despite differences in RCCA insertion, the DNBR is always greater at part power than at full power.

3.1.3.2 Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are set:

1. Sufficient control is available to produce a hot shutdown margin of at least 1 percent $\Delta k/k$.
2. The shutdown margin is maintained with the most reactive RCCA stuck in the fully withdrawn position.
3. The shutdown margin is maintained at ambient temperature by the use of soluble poison.

3.1.3.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

1. The minimum allowable DNBR during normal operation, including anticipated transients, meets the applicable DNBR correlation limit.
2. No fuel melting during any anticipated normal operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a Departure from Nucleate Boiling (DNB) which causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant resulting in high clad temperatures.

The ratio of the heat flux causing DNB at a particular core location, as predicted by the applicable DNBR correlation, to the existing heat flux at the same core location is the DNBR. A DNBR correlation limit corresponding to a 95 percent probability at a 95 percent confidence level that DNB does not occur is chosen as an appropriate margin to DNB for all operating conditions.

DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters, reactor power, reactor coolant temperature and pressure have been related to DNB through the applicable DNBR correlation. Curves presented in the Core Operating Limits Report represent the loci of points of reactor power, reactor coolant pressure and average temperature for which the DNBR is less than the applicable correlation limit. The area of safe operation is the lower average temperatures and higher reactor coolant pressures limited by one specified curve of the reactor power parameter family of curves shown. The parameters used in the development of the curves were checked in the course of plant startup tests.

3.1.3.4 Mechanical Limits

3.1.3.4.1 Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup, steady-state operation with any number of pumps running, and shutdown conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel are limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with RCCAs. Core drop in the event of failure of the normal supports is limited so that the rod cluster control assemblies do not disengage from the fuel assembly guide thimbles.

The structural internals are designed to maintain their functional integrity in the event of a major loss-of-coolant accident (LOCA) and the Design Basis Earthquake. The dynamic loading resulting from the pressure oscillations because of a LOCA is discussed in Section 14.3.3.

3.1.3.4.2 Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow-induced vibrations, earthquakes, reactor pressure, fission-gas pressure, fuel growth, thermal strain, and differential expansion during both steady-state and transient reactor operating conditions have been considered in the design of the fuel rods and fuel assembly. The assembly is also structurally designed to withstand handling and shipping loads prior to irradiation, and to maintain sufficient integrity at the completion of design burnup to permit safe removal from the core and subsequent handling during cooldown, shipment and fuel reprocessing or storage.

The fuel rods are supported at several locations along their length within the fuel assemblies by grid assemblies, which are designed to maintain control of the lateral spacing between the rods throughout the design life of the assemblies. The magnitude of the support loads provided by the grids are established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint of sufficient magnitude to result in buckling or distortion of the rods. The fuel rod cladding is designed to withstand operating pressure loads without collapse or rupture and to maintain encapsulation of the fuel throughout the design life.

3.1.3.4.3 Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in the RCCAs are similar to those used for the fuel rod cladding. The stainless steel cladding is designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included in the RCCA cladding thickness.

Adequate clearance is provided between the absorber rods and the guide thimbles which position the rods within the fuel assemblies so that coolant flow along the length of the absorber rods is sufficient to remove the heat generated without overheating of the absorber cladding. The clearance is also sufficient to compensate for any misalignment between the absorber rods and guide thimbles and to prevent mechanical interference between the rods and guide thimbles under any operating conditions.

3.1.3.4.4 RCC Drive Assembly

The RCC drive assemblies for the full-length rods provide rod cluster control assembly insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various RCC groups. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system malfunction or operator error (Section 14.1).

Also, the RCC drive assemblies for the full-length rods provide a fast insertion rate during a “trip” of the RCCAs, which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system. This rate is based on the results of various reactor transient analyses (Chapter 14), including instrument and control delay times and the amount of reactivity that must be inserted before deceleration of the RCCA occurs.

3.1 References

1. Lamb, J. G., (NRC) to T. Coutu (NMC) transmitting the NRC SER for Amendment No. 168 to the Operating License, approving the Measurement Uncertainty Recapture (MUR) Power Uprate (1.4%), Letter No. K-03-094, July 8, 2003
2. Lamb, J. G., (NRC) to T. Coutu (NMC) transmitting the NRC SER For Amendment No. 172 to the Operating License, approving the 6% Stretch Power Uprate, Letter No. K-035, February 27, 2004

3.2 REACTOR DESIGN

3.2.1 Nuclear Design And Evaluation

This section presents the nuclear characteristics of the core and an evaluation of the characteristics and design parameters which are significant to design objectives. The capability of the reactor to achieve these objectives while performing safely under operational modes, including both transient and steady state, is described.

3.2.1.1 Reactivity Control Aspects

Reactivity control is provided by:

- a soluble chemical neutron absorber in the reactor coolant (boric acid, also called chem shim),
- movable neutron-absorbing RCCAs,
- discrete burnable poison rods (as specified in the design), and
- integral burnable poison material (as specified in the design).

The concentration of boric acid is varied as necessary during the life of the core to compensate for:

- changes in reactivity which occur with change in temperature of the reactor coolant from cold shutdown to the hot operating, zero-power conditions;
- changes in reactivity associated with changes in the fission product poisons xenon and samarium;
- reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium);
- changes in reactivity due to discrete and integral burnable poison depletion; and
- load-follow operation (power defect).

The RCCs provide reactivity control for:

- fast shutdown;
- reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level);
- reactivity associated with any void formation;
- reactivity changes associated with the power coefficient of reactivity; and

- axial offset control.

The RCCs are divided into two categories according to their function. The RCCs, which compensate for changes in reactivity due to variations in operating conditions of the reactor, such as power or temperature, comprise the control group of RCCs. The other RCCs provide additional shutdown reactivity and are termed shutdown RCCs. The total worth of all the RCCs provides adequate shutdown margin with the most reactive RCCA stuck out of the core (Reference Table 3.2-2).

The discrete burnable poison rods and integral burnable poison material provide control of part of the excess reactivity available during the initial portion of the cycle. The primary functions of the burnable poison are to prevent the moderator temperature coefficient from being positive, under normal operating conditions, by reducing the soluble poison content of the core at the beginning of life (Reference 4); and to provide radial power shaping to control power peaking.

3.2.1.2 Nuclear Design Data

3.2.1.2.1 Core Reactivity Characteristics

A summary of nuclear design data including core reactivity characteristics is presented in Table 3.2-1.

Control to render the reactor sub-critical at temperatures below the operating range is provided by the chemical shim. The minimum required boron concentration during refueling as specified by the Technical Specifications, together with the RCCAs, provides at least 5 percent shutdown margin for these operations. This boron concentration is also sufficient to maintain the core sub-critical ($k < 0.99$) without any RCCAs during refueling. The boron concentration for refueling is equivalent to less than 2 percent by weight boric acid (H_3BO_3) and is well within solubility limits at ambient temperature. This concentration is also maintained in the spent fuel pool, since it is directly connected with the reactor refueling cavity during refueling operations.

The typical initial full-power boron concentration with all RCCAs out and without equilibrium xenon and samarium is shown in Table 3.2-1. As the fission product poison concentrations are built up to their equilibrium full power levels, the boron concentration is reduced to the typical value shown in Table 3.2-1.

Extensive administrative controls render the possibility of loading fuel assemblies without their discrete burnable poison rods, in the event that they are used, or control rods extremely unlikely. Independent checks are made, prior to fuel loading, of each fuel assembly, matching the contents of the assembly with its position in the core. Further checks are provided during core loading, utilizing detailed written handling, and check-off procedures. The specific administrative precautions exercised render it unreasonable to assume the cases of a partially or fully loaded core without discrete burnable poisons or control rods, and so these cases are not analyzed. Achieving criticality during the initial core loading is prohibited in any case as the sub-critical neutron flux is continuously monitored and the inverse count rate ratio is plotted (1/m plot) to detect any unexpected rise in the sub-critical neutron flux. Core loading is stopped should the sub-critical count rate rise by more than a preset factor. Any such loading error not significant enough to be detected during core loading is of no consequence from a criticality standpoint and would be detected during in-core flux mapping prior to full power operation of the plant.

3.2.1.2.2 Kinetic Characteristics

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables, which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the Reactor Control and Protection System.

3.2.1.2.3 Moderator Temperature Coefficient of Reactivity

The moderator temperature coefficient (MTC) in a core controlled by chemical shim is less negative or more positive than the MTC in a core without chemical shim control. The chemical shim poison density is decreased with the water density upon an increase in temperature. This gives rise to a positive component of the moderator temperature coefficient due to the boron being removed from the core. This component is directly proportional to the amount of reactivity controlled by the dissolved poison. To reduce the soluble poison requirement for control of excess reactivity, discrete and/or integral burnable poisons are typically incorporated in the core design. The result is that changes in the coolant density will have a reduced positive component to the MTC and the MTC will be more negative or less positive.

The discrete burnable poison for the initial core was in the form of borosilicate glass rods clad in stainless steel. Reload design cores continued to use the standard discrete burnable poison rod design as the main burnable poison through cycle 22. Starting with cycle 23, Gadolinia was used as the main burnable poison. When used, clusters of discrete burnable poison rods are distributed throughout the core in vacant rod cluster control guide thimbles. Information regarding research, development and nuclear evaluation of the discrete burnable poison rods can be found in Reference 1 and Reference 4. Integral burnable poisons take the form of Gadolinia dispersed throughout the fuel pellets in selected fuel rods in selected fuel assemblies. Gadolinia weight percentages of 2, 4, 6, or 8 may be used.

The moderator temperature coefficient becomes more negative with increasing burnup, resulting from buildup of plutonium and fission products and dilution of the boric acid concentration. The range of the calculated unrodded moderator temperature coefficient is shown in Table 3.2-1.

3.2.1.2.4 Moderator Pressure Coefficient of Reactivity

The moderator pressure coefficient is positive at plant operating conditions. Its effect on core reactivity and stability is small because of the small magnitude of the pressure coefficient, a change of 50 psi in pressure having no more effect on reactivity than a half-degree change in moderator temperature.

3.2.1.2.5 Moderator Density Coefficient of Reactivity

A uniform moderator density coefficient is defined as a change in the neutron multiplication per unit change in moderator density. The range of the moderator density coefficient from BOL to EOL is specified in Table 3.2-1.

3.2.1.2.6 Doppler and Power Coefficient of Reactivity

The Doppler Coefficient is defined as the change in reactivity per degree change in fuel temperature. The range of the Doppler Temperature Coefficient is specified in Table 3.2-1.

To obtain the change in reactivity with power (power coefficient) it is necessary to know the change in the effective fuel temperature with power and the Doppler temperature coefficient. Also, the power coefficient includes the reactivity change due to the moderator temperature change with power.

3.2.1.2.7 Shutdown Margin and RCCA Reactivity Requirements

RCCA reactivity and shutdown margin requirements at beginning and end of life for a typical core are summarized in Table 3.2-2.

RCCAs are available to compensate for the reactivity change incurred with a change in power level. The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is a part of the power-dependent reactivity change along with the Doppler effect and void formation, the associated reactivity change is controlled by RCCAs. The largest amount of reactivity that is controlled is at the end of cycle when the moderator temperature coefficient has its most negative value.

3.2.1.2.8 Operational Maneuvering Band

The control rod groups can be operated within a range from fully withdrawn to insertion to the Control Bank Insertion Limits provided that core axial power distribution is maintained within the limits specified by axial flux difference control procedures. The control rod groups can be maintained in the operational maneuvering band by using changes in the reactor coolant boron concentration to compensate for reactivity changes.

3.2.1.2.9 Xenon Stability Control

Out-of-core instrumentation is adequate to enable the operator to monitor and control xenon-induced power oscillations. A full discussion of xenon stability control can be found in Reference 2.

3.2.1.2.10 Reactor Core Power Distribution

In order to meet the performance objectives without violating safety limits, the peak-to-average power density must be within the limits set by the nuclear hot channel factors. For the peak power point in the core at rated power, the nuclear heat flux hot channel factor, F^N_q, must meet the established F^N_q limit required by Technical Specifications. For the hottest channel at rated power the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, must meet the established $F^N_{\Delta H}$ limit required by Technical Specifications.

Power capability of a pressurized water reactor (PWR) core is determined largely by consideration of the power distribution and its interrelationship to limiting conditions involving:

- the linear power density,
- the fuel cladding integrity, and
- the enthalpy rise of the coolant.

To determine the core power capability, local as well as gross core neutron flux distributions have been determined for various operating conditions at different times in core life.

The presence of RCCAs, discrete and integral burnable poisons, and chemical shim concentration all play significant roles in establishing the fission power distribution, in addition to the influence of thermal-hydraulic and temperature feedback considerations.

Thermal-hydraulic feedback considerations are especially important late in cycle life where the magnitude of the flux redistribution and reactivity change with change in core power or RCC movement are strongly influenced by enthalpy rise up the core and by the fuel burnup distribution. Consequently, each reload is evaluated by extensive power distribution analyses, and thermal hydraulic analyses where necessary. These calculations are verified through start-up tests performed as described in the reactor test program (see Reference 5). In-core instrumentation is employed to evaluate the core power distributions throughout core lifetime to assure that the thermal design criteria are met.

Administrative procedures, alarm functions and automatic rod stops guide and monitor the operator in performing these tasks. The out-of-core nuclear instrumentation system supplies the necessary information for the operator to control the core power distribution within the limits established for the protection system design. This information is provided by a recorder and meter for each long ion chamber which displays the upper and lower ion chamber currents and an indicator which gives the difference in these two currents for each long ion chamber. The ion chamber currents to the recorders and indicators are calibrated against in-core power distribution obtained from the movable detector system generated in the adjacent section of the core. This essentially divides the core into eight sections, four in the upper half and four in the lower half.

The relationship between core power distribution and out-of-core nuclear instrumentation readings was established during the startup testing program. In-core flux measurements were made over a range of relative positions between part-length rods[1] and full-length rod control banks for reactor power in the range of 25 percent to 100 percent. These measurements, together with long ion chamber currents, were processed to yield the relationships between core average axial power generation, the axial peaking factor and axial offset as indicated by the out-of-core nuclear instrumentation. In-core to ex-core power distribution relationships are determined each reload cycle for proper system calibration.

The key power-range functions are consistent with the topical report WCAP-7811 (Reference 2) these functions have also been validated by direct measurement at the Ginna reactor (which has the same functions) and reported in the topical report WCAP-7756 (Reference 3). The power range nuclear instrumentation system is used to measure power level, top-to-bottom power balance, and alarm quadrant-to-quadrant power imbalance. Suitable alarms and protective actions are derived from these signals as described below.

Basic power-range signals are:

1. total current from a power-range detector (four such signals from separate detectors); these detectors are vertical and have an active length of 10 feet;
2. current from the upper half of each power-range detector (four such signals); and
3. current from the lower half of each power-range detector (four such signals).

Derived from these basic signals are the following (including standard signal processing for calibration):

4. indicated nuclear power (four such); and
5. indicated axial power imbalance, derived from upper-half power minus lower-half power (four such).

Alarm functions derived are as follows:

6. deviation (maximum minus minimum of four) in indicated nuclear power;
7. upper tilt (maximum-to-average of four) on upper-half/currents; and
8. lower tilt (maximum-to-average of four) on lower-half/currents.

1. Part-length rods have been removed.

Reactor Trip functions derived are as follows:

9. high neutron flux (two of four); and
10. excess axial power imbalance (two of four) through axial power imbalance to the Overtemperature-Delta-T protection feature.

Provision is made to continuously record, on recorders on the vertical panels, the eight half-currents. Nuclear power is selectable for recording as well. Indicators are provided on the control board for nuclear power and for axial power imbalance.

Also, the reactor is operated at or below 100 percent of rating. The indicated nuclear power level is calibrated periodically to match the thermal output of the core, as determined by a secondary side heat balance measurement.

The thermal limits for the core are quantified as a peak power density limit (to assure that fuel does not melt) and by a minimum transient DNBR limit to assure the nucleate boiling heat transfer is maintained. Direct protection for these two thermal limits is assured by the Overpower-Delta-T and Overtemperature-Delta-T protection features respectively. Because a gross imbalance in the axial power distribution may have an impact on peak power density and on DNBR, the axial power imbalance signal is used as an input to the Overtemperature Delta-T protection feature. The original basis for the numerical settings is given in WCAP-7811 (Reference 2) with additional explanatory material in WCAP-7756 (Reference 3). Limiting values for these settings are given in the Technical Specifications.

The reactor core may be subject to axial xenon oscillations at the end of a fuel cycle life. The axial instability is due principally to the negative moderator temperature coefficient of reactivity, which exists at end of life. Since the moderator coefficient at beginning of life is small, stability against axial oscillations is greatly increased at beginning of life. Consequently, stability margin experiments would not be informative at beginning of life.

A more detailed discussion of the background, analytical and experimental data which forms the basis for this approach, is given in WCAP 7811 (Reference 2).

3.2.1.2.11 Nuclear Core Design and Evaluation

The basis for confidence in the core design methods comes from comparisons of the methods with experimental data and actual plant measurements. Measurements for each reload core consist of startup physics test measurements as well as at power core operation power distribution and reactivity measurements.

3.2.2 Thermal and Hydraulic Design and Evaluation

3.2.2.1 Introduction and Summary

This section describes the calculational methods used for the thermal-hydraulic analysis, the DNB performance, and the hydraulic compatibility during the transition from an all Framatome/ANP 14x14 heavy fuel core through mixed-fuel cores to an all Westinghouse 422V+ core.

Full-scale hydraulic tests (Reference 6) were performed on the 14x14 Westinghouse (422V+) fuel assembly design to confirm the hydraulic compatibility with the FRA/ANP (FRA-ANP) 14x14 fuel design. Based on comparison of the overall core pressure loss, the 14x14 Westinghouse 422V+ fuel assembly design pressure drop is approximately 10 percent higher than the FRA/ANP 14x14 design. A comparison of the 14x14 Westinghouse 422V+ fuel assembly design to the FRA/ANP 14x14 design, in terms of fuel assembly crossflow velocities due to grid pressure drop mismatch, was also made. A baseline for this comparison was established by a Westinghouse transition core that successfully transitioned from 14x14 STANDARD fuel design to the 14x14 optimized fuel assembly (OFA) design. The results of the crossflow analyses show that the transition from the FRA/ANP 14x14 fuel to the 14x14 Westinghouse 422V+ fuel assembly design for KNPP is bounded by the Westinghouse transition core operational experience base.

The 422V+ design allows power uprating at the current $F^N_{\Delta H}$ limit of 1.70. Table 3.2-3 presents a comparison between the previous thermal-hydraulic design parameters and the thermal-hydraulic design parameters that were used in this analysis. All of the thermal-hydraulic design criteria are satisfied for the KNPP fuel transition and power uprate to 1772 MWth reactor power with these thermal hydraulic design parameters.

3.2.2.2 Methodology

The thermal-hydraulic analysis of the 14x14 422V+ fuel is based on the Revised Thermal Design Procedure (RTDP) (Reference 7) and the WRB-1 DNB correlation (Reference 8). The DNB analysis of the core containing 14x14 422V+ fuel assemblies has been shown to be valid with the WRB-1 DNB correlation (Reference 8 and Reference 9), RTDP methods and the VIPRE-W Model (Reference 10). The W-3 correlation and Standard Thermal Design Procedure (STDP) are used when any one of the reactor conditions are outside the range of the WRB-1 correlation (that is, pressure, local mass velocity, local quality, heated length, grid spacing, equivalent hydraulic diameter, equivalent heated hydraulic diameter, and distance from last grid to critical heat flux site) and RTDP (that is, the statistical variance is exceeded on power, T_{IN}, pressure, flow, bypass, $F^N_{\Delta H}$, $F^E_{\Delta H,1}$, and F^E_Q).

The WRB-1 DNB correlation is based entirely on rod bundle data and takes credit for the improvements in the accuracy of the critical heat flux predictions over previous DNB correlations. The approval, by the Nuclear Regulatory Commission (NRC), that a 95/95 correlation limit DNBR of 1.17 is appropriate for the 14x14 OFA fuel assemblies has been documented (Reference 3). Furthermore, it has been shown that the use of the WRB-1 correlation with a 95/95 correlation limit DNBR of 1.17 is appropriate for the 14x14 422V+ fuel assemblies.

With RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor. This factor is used to define the design limit DNBR that satisfies the DNB design criterion (that is, a plant-specific design limit is that value that accounts for the RTDP uncertainties above the correlation DNBR limit). The criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent (at 95 percent confidence level) for any Condition I or II event (that is, normal operation or anticipated operational occurrences). Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values. For cases where conditions fall outside the bounds of the RTDP methodology (that is, the statistical variance is exceeded on power, T_{IN}, pressure, flow, bypass, $F^N_{\Delta H}$, $F^E_{\Delta H,1}$, and F^E_Q), STDPs are used and the associated analyses are performed using input parameters with their uncertainties included.

The uncertainties included in the combined peaking factor uncertainty are:

- The nuclear enthalpy rise hot channel factor, ($F^N_{\Delta H}$)
- The enthalpy rise engineering hot channel factor, ($F^E_{\Delta H}$)
- Uncertainties in the VIPRE-W and transient codes
- Uncertainties based on surveillance data associated with vessel coolant flow, core power, coolant temperature, system pressure, and effective core flow fraction (that is, bypass flow).

The increase in DNB margin is realized when nominal values of the peaking and hot channel factors are used in the DNB safety analyses. Table 3.2-4 provides a listing and description of the peaking factor uncertainties.

Instrumentation uncertainties are documented in the KNPP RTDP Instrument Uncertainty Methodology Report (Reference 5). The uncertainties have changed from those previously used for the KNPP analysis to those listed below (such as, power uprating and the current regulatory environment have all been considered in assessing the need to increase the various plant parameter uncertainties). Both the calculated uncertainties and the uncertainties used in the safety analysis, which were used for the fuel upgrade/plant uprating analyses, are listed in Table 3.2-5. The instrumentation uncertainties were used in determining the DNBR design limits. It should be noted that the uncertainties used in safety analysis are slightly larger than what was calculated during the RTDP uncertainty analysis. The rationale of using slightly larger values for the uncertainties ensures conservatism in determining the DNBR design limit and conservatism in the overall analysis.

The design limit DNBR values for the 422V+ fuel are 1.25/1.25 for typical/thimble cells. For use in the DNB safety analyses, the design limit DNBR is conservatively increased to provide DNB margin to offset the effect of rod bow, transition core, and any other DNB penalties that may occur, and to provide flexibility in design and operation of the plant. This increase in the design limit to account for various penalties and operational issues is the plant-specific margin retained between the design limit and the safety analysis limit. After accounting for the plant-specific margin, the safety analysis limit for the 422V+ fuel is 1.34/1.34 (typical/thimble). These safety analysis limits are employed in the DNB analyses.

With the safety analysis limit set, the core limit lines, axial offset limit lines, and dropped rod limit lines are generated. In generating the various limit lines, the maximum $F^N_{\Delta H}$ is determined that yields acceptable results based upon the safety analysis limits. Based on generating these limit lines, the maximum $F^N_{\Delta H}$ limit that can be supported is 1.70 (including uncertainties) for the 422V+ fuel. Included uncertainty that has been accounted for is the measurement uncertainty of 4 percent (Reference 12).

$$F^N_{\Delta H} = 1.70 \text{ x } [1 + 0.3(1\text{-}P)]$$

Where

P = the fraction of full power

Table 3.2-6 summarizes the available DNBR margin for KNPP. It should be noted that the DNBR margin summaries are cycle dependent and may vary from that documented here in future reload designs.

The high thermal performance (HTP) correlation (Reference 19) is used to predict the DNB heat flux for the FRA-ANP fuel containing HTP spacers. The HTP correlation is a linear function of the local quality with coefficients that are polynomial functions of pressure and local coolant mass flux. The correlation is modified by factors, which account for the effects of non-uniform axial power distribution and fuel design parameters. The HTP DNB correlation and associated DNBR limit have been implemented for use with FRA-ANP fuel with HTP spacers. The NRC has reviewed and approved the use of HTP correlation and associated DNBR limit for FRA-ANP fuel with HTP spacers (Reference 20 and Reference 21).

3.2.2.3 Hydraulic Compatibility

The 14x14 422V+ and 14x14 FRA/ANP fuel assembly designs have been shown to be hydraulically compatible (Reference 1), based on a consistent comparison of the component loss coefficients, thus minimizing effects of fuel assembly crossflow. The axial grid locations, grid heights, and fuel assembly pitch and envelope for the 14x14 422V+ are consistent with the 14x14 FRA/ANP design, again minimizing assembly-to-assembly crossflow. By maintaining grid-to-grid overlap between the FRA/ANP design and the 422V+ design, excessive crossflow between assemblies is prevented. The small difference in loss coefficients between the two designs and the respective grid locations of the two designs has been analyzed to demonstrate that no crossflow-induced vibration will result in a condition in which fretting or whirling would be induced. The fuel assembly crossflow that exists for the transition core is well within the bounding Westinghouse experience basis of transition core analysis (that is, transition cores with intermediate flow mixing vane grids will experience the maximum crossflow situation).

A second area of hydraulic compatibility associated with higher resistance fuel assemblies (the 422V+ design) is the associated impact on lift forces. When a fuel assembly with a different hydraulic resistance is loaded into a core, it changes the flow distribution in the surrounding assemblies. In particular, if this fuel assembly has a higher value of fuel assembly loss coefficient, the surrounding assemblies (that is, lower resistance fuel assemblies - the FRA/ANP assemblies) would see a higher average flow through them than they would in a full core situation. Thus, the lift force on these surrounding assemblies can be expected to increase. The larger the number is of high resistance fuel assemblies loaded in the core, the greater the lift force is on the lower resistance assemblies.

The 14x14 Westinghouse 422V+ fuel assembly design overall pressure loss is approximately 10 percent larger than the FRA/ANP 14x14 fuel (Reference 6). The 10 percent increase in pressure loss will equate to a 10 percent increase in the fuel assembly lift force. When a majority of the core is loaded with Westinghouse fuel, the FRA/ANP fuel can experience, in the limiting situation, a fuel assembly pressure loss equal to the Westinghouse fuel design or an increase of 10 percent in lift force.

3.2.2.4 Effects Of Fuel Rod Bow On DNBR

The concern with regards to fuel rod bow phenomenon is the potential effects on fuel assembly power distribution and on the margin to DNB. Thus, the phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events. Fuel rod bow is the phenomenon of fuel rods bowing between mid-grids. The effect of the rod bow is to impact the channel spacing between adjacent fuel rods. With a reduced channel spacing, the potential of DNB occurring increases. To determine the impact of rod bow on DNB, Westinghouse conducted tests to determine the impact of rod bow on DNB performance. These tests and subsequent analyses were documented in Reference 13. Currently, the maximum rod bow penalty for the OFA fuel assembly is 2.6 percent DNBR at an assembly average burnup of 24,000 MWD/MTU (Reference 13 and Reference 14). For burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory (Reference 15). Therefore, no additional rod bow penalty is required at burnups greater than 24,000 MWD/MTU. Based on the testing and analyses of various fuel designs (Reference 13), including the 14x14 STANDARD, evaluations have shown that the 14x14 OFA and the 14x14 422V+ fuel assemblies will have the same rod bow penalty applied to the analysis basis as that used for 14x14 STANDARD fuel assemblies. For the 422V+ application, the rod bow penalty will be offset with DNB margin retained between the safety analysis and design DNBR limits (refer to Table 3.2-5).

3.2.2.5 Fuel Temperature/Pressure Analysis

Fuel temperatures and associated rod internal pressures have been generated (Reference 16) for the 422V+ fuel. The characteristics of the Gd fuel are such that the Gd rods would exhibit higher fuel temperatures due to an inherent lower thermal conductivity of the Gd-bearing fuel pellet. In addition, increasing Gadolinia enrichment results in a corresponding decrease in the fuel melting temperature. The performance criteria employed by for Gd rods is to ensure that these rods are less limiting than the non-Gd rods, throughout life, in terms of fuel temperatures, rod internal pressures, and core stored energy. This is achieved by holding down the U^{235} enrichment in the Gd rods so that the Gd rods are at sufficiently lower power throughout life. Therefore, the fuel performance parameters for the 422V+ fuel bound those for the 422V+ Gd fuel. The higher fuel rod average and surface temperatures are conservative for the accident and transient analyses.

Fuel centerline temperatures were also generated for the 422V+ fuel. These are used for verification that fuel melt will not occur. The maximum kW/ft limit for fuel melt is 22.54 kW/ft for the 422V+ fuel.

In addition to the fuel temperatures and pressures, the revised core stored energy for the 422V+ fuel has been determined for use in containment analysis. Core stored energy is defined as the amount of energy in the fuel rods in the core above the local coolant temperature. The local core stored energy is normalized to the local linear power level. The units for the core stored energy are in full-power seconds (FPS). The value of the core stored energy for the 422V+ fuel design is 4.68 FPS.

3.2.2.6 Transition Core Effect

Redistribution of flow in PWR cores is a well documented and modeled phenomenon that occurs generally because of thermal-hydraulic fluid condition gradients within the core. In a mixed core, with assemblies having different hydraulic resistance, the local hydraulic resistance differences are also a mechanism for flow redistribution. This redistribution results in the fluid velocity vector having a lateral component as well as the dominant axial component. The lateral component is commonly referred to as crossflow. The crossflow induced by local hydraulic resistance differences will typically impact the mechanical design of the fuel assemblies, as well as the safety analyses of the core.

The mechanical design of the fuel assemblies in the core could be affected in two ways:

1. Excitation of peripheral rods in the fuel assemblies such that wear mechanisms of fretting or whirling could exist, and
2. Introduction of higher resistance assemblies will influence the lift forces on the remaining assemblies. The hydraulic compatibility of the 422V+ and FRA/ANP fuel assemblies has been addressed and found to be acceptable.

In the safety analysis, crossflow affects both LOCA and DNB. Primary consideration for the LOCA analysis is the reduction of the normalized mass velocity as compared to a full core of that assembly type. DNB is affected because the flow redistribution affects both mass velocity and enthalpy distributions. With current DNB correlations, WRB-1 and W-3, the flow redistribution occurs at the location that the minimum DNBR is predicted. As such, the design verification is based on the principle that once the transition core DNBR penalty is determined, all further plant-specific analysis may proceed as if it were a full core analysis.

Transition cores are analyzed as if they were full cores of one assembly type (full 422V+) and applying the transition core penalty. Penalties are a function of the number of each type of fuel assembly in the core (Reference 17), which has been approved by the NRC (Reference 18). This methodology is used to calculate the FRA/ANP to 422V+ transition core penalties. There is no DNBR transition core penalty for the FRA/ANP fuel. However, less than 3.0 percent DNBR penalty applies to the first cycle 422V+ fuel for the first transition cycle operation. The penalty is applied follows:

The 422V+ penalty starts at 4.22 percent (with only one 422V+ assembly in the core) and decreases to a minimum value per the following equation as the number of 422V+ fuel assemblies is increased. The minimum 422V+ penalty is zero with an all 422V+ core. (In actuality, the penalty applied to the 422V+ fuel would be on the order of 2.53 percent with one-third of the core being 422V+ fuel - first transition cycle. The minimum penalty would occur in the second transition cycle where two-thirds of the core is 422V+ fuel. This minimum penalty would be on the order of 1.0 percent.) A 95/95 uncertainty on the transition core penalty of 0.30 percent must be added to the transition core penalty for the 422V+ fuel.

DNBR Penalty 422V + (%) = -5.07 x 4.22

The value of x in the above equation is the fraction of that type of fuel remaining in the core. The penalty applied to the 422V+ fuel is due primarily to the slightly higher flow resistance of the 422V+ design.

Bypass Flow

Two different bypass flow rates are used in the thermal-hydraulic design analysis: thermal design bypass flow (TDBF) and best-estimate bypass flow (BEBF). These two bypass flows are used in non-statistical and statistical analyses respectively. The TDBF is the conservatively high core bypass flow used in calculations where the results are adversely affected by low core flow. Specifically, TDBF is used with the vessel thermal design flow (TDF) in power capability analyses that use standard (non-statistical) methods. The TDBF is also used with the vessel best-estimate flow (BEF) to calculate core and fuel assembly pressure drops. The BEBF is the flow that would be expected using nominal values for dimensions and operating parameters that affect bypass flow without applying any uncertainty factors. The BEBF is used in conjunction with the vessel minimum measured flow for power capability analyses that use improved thermal design procedure or RTDP (statistical methodology). It is also used to calculate fuel assembly lift forces. For the KNPP analyses, the maximum permissible TDBF is 7.0 percent and the maximum permissible BEBF is 5.5 percent.

3.2.2.7 Thermal-Hydraulic Design Parameters

Table 3.2-3 lists numerous thermal-hydraulic parameters for the current rated reactor power of 1772 MWt with 422V+ fuel. The following parameters from Table 3.2-3 are used in the VIPRE-W model:

- Reactor core heat output (MWt)
- Core pressure for RTDP analyses (psia)
- Heat generated in fuel (%)
- Average heat flux (BTU/hr-ft^2)
- Nominal vessel/core inlet temperature (°F)
- $F^N_{\Delta H}$, nuclear enthalpy rise hot-channel factor
- Pressurizer/core pressure (psia)
- Thermal design flow for non-RTDP analyses (gpm)
- Minimum measured flow for RTDP analyses (gpm)

In addition, the average linear power (kW/ft) is used in the analyses for the fuel temperatures and other fuel rod design criteria. The limiting direction for these parameters is as shown in Table 3.2-7.

3.2.2.8 Conclusion

The thermal-hydraulic evaluation of the fuel transition for KNPP has shown that FRA/ANP and 422V+ fuel assemblies are hydraulically compatible and that the DNB margin gained through use of the RTDP methodology with the WRB-1 DNB correlation is sufficient to allow an increase in the power to 1772 MWt. More than sufficient DNBR margin in the safety limit DNBR exists to cover any rod bow and transition core penalties. All current thermal-hydraulic design criteria are satisfied.

3.2.3 Mechanical Design and Evaluation

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2-2 and in elevation in Figure 3.2-3. The core, consisting of the fuel assemblies with and without integral poison, RCCAs, and discrete burnable poison rods (if used) provides and controls the heat source for the reactor operation. The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation. A listing of the core mechanical design parameters is given in Table 3.2-8.

The fuel assemblies are arranged in a roughly circular cross-sectional pattern. The assemblies are all mechanically compatible, but contain fuel of different enrichments depending on the location of the assembly within the core.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. Pellets in selected fuel rods may contain integral Gadolinia burnable poison. The pellets are stacked to an active height of 144 inches (143.25 inches for the Westinghouse 422V+ lead use assemblies first inserted for cycle 25) within Zircaloy-4 tubular cladding, (ZIRLO for the Westinghouse 422V+ lead use assemblies first inserted for cycle 25) which is plugged and seal-welded at the ends to encapsulate the fuel. All fuel rods are internally pressurized with helium during fabrication. Heat generated by the fuel is removed by de-mineralized borated light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The control rods, designated as RCCAs, consist of groups of individual absorber rods, which are held together by a spider at the top end and actuated as a group. In the inserted position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes, which form an integral part of the upper core support structure. Figure 3.2-4 shows a typical rod cluster control assembly.

As shown in Figure 3.2-3, the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. The core plates are provided with pins, which index into closely fitting mating holes in the fuel assembly top and bottom nozzles. The pins maintain the fuel assembly alignment, which permits free movement of the control rods from the fuel assembly into the guide tubes in the upper support structure without binding or restriction between the rods and their guide surfaces.

Operational or seismic loads imposed on the fuel assemblies are transmitted through the core plates to the upper and lower support structures and ultimately to the internals support ledge at the pressure vessel flange in the case of vertical loads or to the lower radial support and internals support ledge in the case of horizontal loads. The internals also provide a formfitting baffle surrounding the fuel assemblies, which confines the upward flow of coolant in the core area to the fuel-bearing region.

3.2.3.1 Reactor Internals

3.2.3.1.1 Design Description

The reactor internals are designed to support and orient the reactor core fuel assemblies and RCCA, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel, and support in-core instrumentation. The reactor internals are shown in Figure 3.2-3.

The internals are designed to withstand the forces due to weight, pre-load of fuel assemblies, RCCA dynamic loading, vibration, possible blowdown forces, and earthquake acceleration. These internals were analyzed in a manner similar to Connecticut Yankee, San Onofre, Zorita, Saxton, and Yankee. Under the loading conditions, including conservative effects of design earthquake loading, the structure satisfies stress values prescribed in Section III, ASME Nuclear Vessel Code. The dynamic criteria for design and stress levels of the internals in this plant, similar to those in Connecticut Yankee, are described in Appendix B.

The reactor internals are equipped with bottom-mounted in-core instrumentation supports. These supports are designed to sustain the applicable loads outlined above.

In the event of downward vertical displacement of the internals, energy-absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy absorbing devices to the vessel. The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design are determined so as to limit the forces imposed to a safe fraction of yield strength. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy-absorbing devices. See Figure 3.2-5.

The free fall in the hot condition is on the order of 2 inch, and there is an additional strain displacement in the energy absorbing devices of approximately ¾ inch. Alignment features in the internals prevent tilting of the internals structure during this postulated drop. The control system as designed provides assurance of control rod insertion capabilities under these assumed drop conditions. The drop distance of about 1¼ inch is not enough to cause the tips of the shutdown group of RCC assemblies to come out of the guide tubes in the fuel assemblies.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure, and the in-core instrumentation support structure.

Additional flexibility was incorporated into the Kewaunee internals by providing additional holes in the upper support plate and upper core plate that could accommodate additional guide tubes. These holes are identical to those on R.E. Ginna, the two-loop prototype, and were initially covered with cover plates. The holes were restricted with orifice plates in order to give the hole the same pressure drop as any other hole in the upper core plate.

3.2.3.1.2 Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.2-5. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, the intermediate diffuser plate and the core support forging which is welded to the core barrel. All the major material for this structure is Type 304 Stainless Steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates, which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate provides the necessary flow distributor holes for each fuel assembly. Fuel assembly locating-pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support forging of the core barrel in order to provide stiffness to this plate and transmit the core load to the core support forging. Intermediate between the core support forging and lower core support plate is a perforated plate positioned to uniformly diffuse the coolant flowing into the core.

Irradiation baskets in which material samples can be inserted and irradiated during reactor operation are attached to the thermal shield. The irradiation capsule basket supports are welded to the thermal shield. There is no extension of this support above the thermal shield as was done in the older designs. Thus, the basket has been removed from the high flow disturbance zone. The welded attachment to the shield extends the full length of the support except for small interruptions about 1 inch long. This type of attachment has an extremely high natural frequency. The specimens are held in position within the baskets by a stop on the bottom and a slotted cylindrical spring at the top, which fits against a relief in the basket. The specimen does not extend through the top of the basket and thus is protected by the basket from the flow.

The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the core support forging, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core plate are arranged to give a very uniform entrance flow distribution to the core. After passing through the core the coolant enters the area of the upper support structure and then flows generally radically to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum, and exits through the vessel outlet nozzles via the control rod guide tubes.

Vertically downward loads from weight, fuel assembly pre-load, RCCA dynamic loading and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support forging and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be shared by the lower radial support to the vessel head flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core support plate to the barrel wall and by a radial support type connection of the upper core plate to slab-sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by “key” and “keyway” joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block is welded to the vessel inner face. Another Inconel block is bolted to each of these blocks, and has a “keyway” geometry. Opposite each of these is a “key” which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the farthest extremity and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Boiler and Pressure Vessel Code limits.

3.2.3.1.3 Upper Core Support Assembly

The upper core support assembly, shown in Figure 3.2-6 consists of the top support plate, deep beam sections and upper core plate between, which are contained support columns and guide tube assemblies. The support columns establish the spacing between the upper support plate, deep beam sections, and the upper core plate and are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies, shown on Figure 3.2-7, sheath and guide the control rod drive shafts and control rods and provide no other mechanical functions. They are fastened to the upper support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the RCCA drive shafts is provided by the RCCA shroud tube which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during the refueling operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at equal angular positions.

Slots are milled into the upper core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating-pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and RCCAs is thereby assured by this system of locating pins in this guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly, and is compressed by the reactor vessel head flange.

Vertical loads from fuel assembly pre-load are transmitted through the upper core plate via the support columns to the deep beams and upper support plate and also through the circumferential spring to the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

3.2.3.1.4 In-Core Instrumentation Support Structures

The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate.

The thermocouples are routed through these port columns and across the upper support plate to positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

3.2.3.1.5 Evaluation of Core Barrel and Thermal Shield

The internals design is based on analysis, test and operational information. Experience in previous Westinghouse PWRs has been evaluated and information derived has been considered in this design. For example, now Westinghouse uses a one-piece thermal shield, which is attached rigidly to the core barrel at one end and flexured at the other. The early designs that malfunctioned were multi-piece thermal shields that rested on vessel lugs and were not rigidly attached at the top.

The Connecticut Yankee reactor and the Zorita reactor core barrels are of the same construction as the Kewaunee reactor core barrel. Deflection measuring devices employed in the Connecticut Yankee reactor during the hot-functional test, and deflection and strain gages employed in the Zorita reactor during the hot-functional test have provided important information that has been used in the design of the present day internals. When the Connecticut Yankee thermal shield was modified to the same design as Southern California Edison, it, too, operated satisfactorily as was evidenced by the examination after the hot-functional test. After these hot-functional tests on all of these reactors, a careful inspection of the internals was provided. All the main structural welds were examined, nozzle interfaces were examined for any differential movement, upper core plate inside supports were examined, the thermal shield attachments to the core barrel including all lockwelds on the devices used to lock the bolts were checked, and no malfunctions were found.

Substantial scale model testing was performed at Westinghouse Atomic Power Division. This included tests, which involved a complete full-scale fuel assembly, which was operated at reactor flow, temperature and pressure conditions. Tests were run on a 1/7th-scale model of the Indian Point 1 reactor. Measurements taken from these tests indicate very little shield movement, on the order of a few mils when scaled up to actual size. Strain gage measurements taken on the core barrel also indicate very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances have been included.

Information gathered from these tests was used in the design of the thermal shield and core barrel. It can be concluded from the testing program and the analyses with the experience gained that the design as employed at Kewaunee Nuclear Power Plant is adequate.

Various combinations of acceptable number and distribution of baffle-barrel-boltings have been evaluated by the Westinghouse Electric Company (Reference 26). An input to the baffle-barrel-bolting evaluation is the dynamic loads postulated by a break in the reactor coolant pressure boundary and branch piping. To reduce the dynamic load on the baffle plates several leak-before-break evaluations were performed for the reactor coolant pressure boundary piping. Additional information for the leak-before-break analysis is found in USAR Section 4.1.3 under reactor coolant pressure boundary rapid propagation failure prevention.

3.2.3.2 Core Components

3.2.3.2.1 Design Descriptions

3.2.3.2.1.1 Fuel Assembly Design Westinghouse fuel assemblies were used in early core life. Currently, FRA-ANP and Westinghouse fuel designs are used in the Kewaunee reactor. All of the FRA-ANP fuel assemblies in the core are of FRA-ANP Heavy design, with the exception of four Westinghouse 422V+ lead use assemblies first inserted in cycle 25 and 44 Westinghouse 422V+ assemblies. A previous FRA-ANP design identified as FRA-ANP Standard was in the Kewaunee core through cycle 24. The FRA-ANP Standard design is similar to the FRA-ANP Heavy design. The Westinghouse 422V+ assemblies are also similar to the FRA-ANP Heavy design. Therefore, the following discussion applies to all three designs. Table 3.2-9 displays information related to the three designs. The axial and radial configuration of a FRA-ANP Heavy fuel assembly is shown in Figure 3.2-8 and Figure 3.2-9.

The fuel rods in the three fuel assembly designs are arranged in a square array with 14-rod locations per side and a nominal centerline-to-centerline pitch of 0.556 inch between rods. Of the total possible 196-rod locations per assembly, 16 are occupied by guide thimbles for the RCC rods or burnable poison rods and one for in-core instrumentation. The remaining 179 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, 7 grid assemblies, 16 absorber rod guide thimbles, and 1 instrumentation thimble.

For the three designs, the guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The top and bottom ends of the guide thimbles are fastened to the top and bottom nozzles, respectively. The grid assemblies, in turn, are fastened to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal frame-work the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

The main differences between the FRA-ANP Standard compared to the FRA-ANP Heavy fuel designs are as follows:

1. The FRA-ANP Heavy design has a slightly larger fuel pellet diameter, resulting in a heavier uranium loading.
2. The FRA-ANP Heavy design has slightly thinner fuel rod cladding to accommodate the larger diameter fuel pellets.
3. The FRA-ANP Heavy design utilizes five high thermal performance (HTP) grids between the top and bottom grids to improve DNB performance. The FRA-ANP Standard design used seven bi-metallic grids. (Note that the cycle 22 reload of FRA-ANP Standard fuel built for Kewaunee used five HTP grids).

4. The FRA-ANP Heavy design utilizes a FUELGUARD^{TM} bottom nozzle design to increase resistance to debris related fuel failures. The Siemens Power Corporation (SPC) Standard design used the conventional SPC bottom nozzle design. (Note that the cycle 22 reload of SPC Standard fuel built for Kewaunee used the FUELGUARD^{TM} bottom nozzle design).

The Westinghouse 422V+ lead use assemblies first inserted in cycle 25 are similar to the FRA-ANP Heavy design, except that the bottom nozzle design is similar to the FRA-ANP Standard design and ZIRLO is used for the cladding material.

3.2.3.2.2 Fuel Assembly Framatome ANP (FRA-ANP) Designs

The overall configuration and the spacing of the fuel assembly are similar in design to the initial core fuel assembly design and are shown in Figure 3.2-8 and Figure 3.2-9 for FRA-ANP Heavy design. The FRA-ANP Standard fuel rod design has 30-mil wall thickness Zircaloy-4 cladding. The FRA-ANP Heavy fuel rod design has 25-mil thickness Zircaloy-4 cladding. The fuel pellets are dished with a mean fuel density approximately 95 percent of theoretical for both designs. The bi-metallic top and bottom spacers are a Zircaloy-4 structure with Inconel 718 springs for both designs. The fuel assembly upper tie plate for both designs is mechanically locked to the Zircaloy-4 guide tubes and may be removed to allow inspection of irradiated fuel rods and the capability to replace a fuel rod.

The FRA-ANP fuel design parameters for the Standard and Heavy designs are shown in Table 3.2-9.

The various components making up the fuel assembly serve the same function as described below for the original Westinghouse design.

3.2.3.2.3 Fuel Assembly Westinghouse 422V+ and Lead Use Design

The Westinghouse 422V+ lead use assemblies are similar to the FRA-ANP Heavy design except that the bottom nozzle design is similar to the FRA-ANP Standard design and ZIRLO is used for the cladding material. The Westinghouse 422V+ lead use assembly fuel design parameters are shown in Table 3.2-9.

3.2.3.2.4 Bottom Nozzle

The bottom nozzle is a square pedestal structure, which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross-section, is fabricated from 304 stainless steel parts consisting of a perforated plate, four angle legs, and four pads or feet. The legs form a plenum space for the inlet coolant to the fuel assembly. The perforated plate serves as the bottom end support for the fuel rods. The bottom support surface for the fuel assembly is formed under the plenum space by the four pads, which are attached to the corner angles.

The most recent FRA-ANP bottom nozzle design includes the FRA-ANP FUELGUARD debris-resistant filter. The FRA-ANP FUELGUARD design employs parallel rows of curved blades to make an effective debris filter while maintaining minimal flow resistance. The FUELGUARD design is shown in Figure 3.2-11. The FRA-ANP Standard and Westinghouse 422V+ designs employ the perforated plate concept described in the preceding paragraph.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle upward to the interior of the fuel assembly through the flow channels in the nozzle plate and to the channel between assemblies.

The RCC guide thimbles, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle end plate. These loads as well as the weight of the assembly are distributed through the nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads, which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating-pins.

3.2.3.2.5 Top Nozzle

The top nozzle is a box-like structure, which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is directed toward the flow holes in the upper core plate. The top nozzle for all three designs is made of stainless steel with four Inconel hold-down springs attached.

The top nozzle is provided with perforations to provide for coolant flow and provides a means of evenly distributing any axial loads imposed on the fuel assemblies among the guide thimbles.

Two pads containing axial through-holes, which are located on diametrically opposite corners of the top nozzle provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle. The springs are fastened in pairs to the top nozzle at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the nozzle. Fastening of each pair of springs is accomplished with a clamp, which fits over the ends of the springs and bolts which pass through the clamp and spring, and thread into the top nozzle. At assembly, the spring mounting bolts are torqued sufficiently to pre-load against the maximum spring load and then lockwelded to the clamp, which is counter-bored to receive the bolt head.

The spring load is obtained through deflection of the spring set by the upper core plate. The spring form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring is bent downward and captured in a key slot in the top plate. The free end of the lower spring is captured by the bent-down leg of the upper spring. This is done to guard against loose parts in the reactor in the unlikely event of spring fracture. In addition, the fit between the upper spring and key slot and between the spring set and the mating slot in the clamp are sized to prevent rotation of either end of the spring set into the control rod path in the event of spring fracture.

In addition to its plenum and structural functions, the nozzle provides a protective housing for components, which mate with the fuel assembly. In handling a fuel assembly with a control rod inserted, the control rod spider is contained within the nozzle. During operation in the reactor, the nozzle protects the absorber rods from coolant cross flows in the unsupported span between the bottom plate of the top nozzle and the end of the guide tube in the upper internals package. Plugging devices[1], which fill the ends of the fuel assembly thimble tubes at unrodded core locations, and the spiders which support the source rods and burnable poison rods are all contained within the fuel top nozzle.

3.2.3.2.6 Guide Thimbles

The RCC guide thimbles in the fuel assembly provide guided channels for the absorber rods during insertion and withdrawal of the RCCAs. They are fabricated from a single piece of Zircaloy-4 tubing, which is drawn to two different diameters. The larger inside diameter at the top provides a relatively large annular area for rapid insertion during a reactor trip and to accommodate a small amount of upward cooling flow during normal operation. The bottom portion of the guide thimble is of reduced diameter to produce a dashpot action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

Flow holes are provided just above the transition of the two diameters to permit the entrance of cooling water during normal operation, and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is closed at the bottom by means of a welded end plug. The end plug is fastened to the bottom nozzle during fuel assembly fabrication.

3.2.3.2.7 Grids

The spring clip grid assemblies consist of individual slotted straps, which are assembled and interlocked in an "egg-crate" type arrangement to permanently join the straps at their points of intersection. Details such as spring fingers, support dimples, mixing vanes and tabs are punched and formed in the individual straps prior to assembly.

Two types of grids are used in the fuel assembly designs. One type of grid has mixing vanes, which project from the edges of the straps into the coolant stream; this grid is used in the high heat region of the fuel assemblies to promote mixing of the coolant. The other type of grid, located at the bottom and top ends of the FRA-ANP design assemblies and in all locations for the

1. The use of these devices is optional and dependent upon the specifics of the reload core design.

Westinghouse 422V+ lead use assemblies is of the non-mixing type. They are similar to the mixing type with the exception that they contain no mixing vanes on the internal straps.

The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or loading and unloading the core. Additional small tabs on the outside straps and the irregular contour of the straps are also for this purpose.

Fuel loadings of FRA-ANP Standard fuel prior to cycle 23 used Zircaloy in the grid material with Inconel spring inserts for all seven grids. The top and bottom grids do not have the mixing vanes used in the five middle grids. The cycle 23 reload of FRA-ANP Standard fuel and the FRA-ANP Heavy fuel used non-vaned bi-metallic top and bottom grids. However, the five middle grids used with these assemblies are HTP spacers. The HTP spacer is an all Zircaloy design, which includes built-in flow channels for enhanced coolant mixing while maintaining a low flow resistance. The HTP design yields an improvement in DNB margin compared to other spacer designs. The spacer is formed by an interlocking rectangular grid of pairs of Zircaloy-4 strips. Pairs of strips with integral rod support channels are welded back-to-back and arranged such that there are four channels supporting each rod. The channels create flow paths, which are slanted at their outlet, causing a vortex flow pattern, which results in increased mixing of the coolant and higher local heat transfer. A HTP spacer is shown in Figure 3.2-12. The Westinghouse 422V+ lead use assemblies first inserted for cycle 25 have top and bottom grids made of Inconel with five intermediate grids made of ZIRLO.

3.2.3.2.8 Fuel Rods

The fuel rods of the FRA-ANP Standard, the FRA-ANP Heavy and the Westinghouse 422V+ designs consist of uranium dioxide ceramic pellets in a slightly cold-worked and partially annealed Zircaloy-4 tubing (ZIRLO for the Westinghouse 422V+ design) which is plugged and seal-welded at the ends to encapsulate the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel swelling due to accumulated fission products without overstressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a carbon steel helical compression spring, which bears on the top of the fuel.

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process. A holddown force in excess of the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack.

The fuel pellets are right circular cylinders consisting of slightly enriched uranium-dioxide powder, which has been compacted by cold-pressing and then sintered to the required density.

The pellets in fuel rods for selected fuel assemblies may have the burnable absorber Gadolinia dispersed within the pellets. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length. The FRA-ANP fuel rod is shown in Figure 3.2-10. Figure 3.2-10 applies to both the FRA-ANP Standard and FRA-ANP Heavy designs. It also applies to the Westinghouse 422V+ design, except that the dimensions differ as shown in Table 3.2-9.

Allowable fuel enrichment limits are delineated in Reference 23.

The manufacturing process and quality control are described in Section 3.3.

3.2.3.2.9 Rod Cluster Control Assemblies

The control rods or RCCAs each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies one of which is shown in Figure 3.2-4 are provided to control the reactivity of the core under operating conditions.

The absorber material used in the control rods is a silver-indium-cadmium alloy, which is essentially “black” to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The absorber material in the form of extruded single length rods is sealed in stainless steel to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip of the absorber rods remains engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the RCCA are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive shaft, are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the RCCA and absorb the impact energy at the end of a trip insertion. The radial vanes are joined to the hub, and the fingers are joined to the vanes by furnace-brazing. A centerpost, which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from Type 304 stainless steel except for the springs, which are Inconel X-750 alloy and the retainer, which is of 17-4 pH material.

The absorber rods are secured to the spider so as to assure trouble-free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

In construction, the silver-indium-cadmium rods are inserted into cold-worked stainless steel tubing, which is then sealed at the bottom and the top by welded end plugs. Sufficient diametral and end clearance, are provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions. Rods of similar design have been successfully used in the Saxton, SELNI, and Indian Point 1 reactors.

3.2.3.2.10 Neutron Source Assemblies

Two neutron source assemblies were utilized in the core and were subsequently replaced during reload Cycles 4 and 5 with a pair of secondary sources. The initial assemblies each contained one combination primary-secondary source rod and three secondary source rods. Neutron source assemblies were removed from the core in 1992 and have not been used in subsequent cycles.

3.2.3.2.11 Plugging Devices

When necessary to limit bypass flow through the RCC guide thimbles in fuel assemblies, which do not contain either control rods, source assemblies, or burnable poison rods, the fuel assemblies at those locations may be fitted with plugging devices. The plugging devices consist of a flat spider plate with short rods suspended from the bottom surface and a spring pack assembly attached to the top surface. The plugging devices fit within the fuel assembly top nozzles. The short rods project into the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack is compressed by the upper core plate when the upper internals package is lowered into place.

All components in the plugging device, except for the springs, are constructed from type 304 stainless steel. The springs (one per plugging device) are wound from an age-hardenable nickel-base alloy to obtain higher strength. Thimble plugging devices were removed from the reactor in Cycle 14 (1988) and have not been used in subsequent cycles.

3.2.3.2.12 Discrete Burnable Poison Rods

Discrete burnable poison rods were employed during the first core cycle and are periodically used as required by the reload core design. The burnable poison rods are statically suspended and positioned in vacant RCC thimble tubes within the fuel assemblies at nonrodded core locations. The poison rods used were of two types, the standard design and the wet annular burnable absorber (WABA) design. The WABA poison rods were removed from the reactor after one fuel cycle and have not been used in subsequent cycles.

The standard design burnable poison rods consist of borosilicate glass tubes contained within type-304 stainless steel tubular cladding, which is plugged and seal-welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin-wall type-304 stainless steel tubular inner liner. A standard burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2-13.

The standard design burnable poison rods are designed in accordance with the standard fuel rod design criteria. The cladding is free-standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass as a result of the B10 (n, a) reaction. The large void volume required for the helium is obtained through the use of glass in tubular form, which provides a central void along the length of the rods. The resulting clad stresses at temperature and pressure are given in WCAP 7113 (Reference 22).

3.2.3.3 Evaluation of Core Components

3.2.3.3.1 Fuel Rod Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods were analyzed by fuel rod computer models (Reference 24 and Reference 25). Data from these analyses was used to calculate geometry changes and creep deformation of the fuel cladding as a function of time. The increase of internal pressure in the fuel rod due to these phenomena is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum.

The maximum allowable strain in the cladding, considering the combined effects of internal fission gas pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than 1 percent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, elevated temperature tensile testing of fuel tubes, dimensional inspection, x-ray of both end and plugwelds, ultrasonic testing and helium-leak test.

In the event of cladding defects, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those, which are gaseous or highly volatile. This retentiveness decreases with increasing temperature or fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

The integrity of fuel rod cladding so as to retain fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits (cladding perforation) in terms of stress and strain were shown to be met in Reference 24 and Reference 25.

For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margin exists between actual operating conditions and the damage limits.

The other parameters having influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   The internal gas pressure required to produce cladding stresses equal to the damage limit under operating conditions is well in excess of the maximum design pressure. The maximum internal pressure occurs at end of life. This internal pressure depends upon the initial pressure, void volume, and fuel rod power history; however, it does not exceed the design limit.

2. Cladding temperature:

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F.

3. Burnup:

   Fuel burnup results in fuel swelling which produces cladding strain. For FRA-ANP fabricated fuel and Westinghouse fabricated fuel, the base fuel burnup limits have been established at 59,000 MWD/MTU maximum assembly exposure and 62,000 MWD/MTU maximum rod exposure. These limits are verified for each reload and may be reduced depending on the reload core design. The design equilibrium batch average discharge burnup is approximately 48,000 MWD/MTU.

4. Fuel temperature and kW/ft:

   At zero burnup, cladding damage for fuel rods has been calculated to occur at about 30 kW/ft based upon cladding strain reaching the damage limit. At this power rating, and neglecting uncertainties in the evaluation of the maximum pellet temperature, 17 percent of the pellet central region is expected to be in the molten condition. The maximum thermal output at rated power was 17.3 kW/ft at time of initial licensing. Current ECCS requirements limit thermal output to 14.47 kW/ft for FRA-ANP Standard fuel, 14.92 kW/ft for FRA-ANP Heavy fuel and 13.70 kW/ft for the Westinghouse 422V+ lead use assemblies first inserted in cycle 25. The 422V+ fuel peak linear power for prevention of centerline melt analysis value is 22.54 kw/ft.

3.2.3.3.2 Evaluation of Burnable Poison Rods

The burnable poison rods, if used, are positively positioned in the core inside RCCA guide thimbles and held down in place by attachment to a plate assembly compressed beneath the upper core plate and hence cannot be the source of any reactivity transient. Due to the low heat generation rate, and the conservative design of the poison rods, there is no possibility for release of the poison as a result of helium pressure or clad heating during accident transients including loss of coolant.

3.2.3.3.3 Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring fingers, RCC control rods, and burnable poison rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for an infinite number of cycles.

In the case of a typical fuel grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small (~ 0.001), and the stress associated with the motion is significantly small (< 100 psi). Likewise, the reactions at the grid spring due to the motion is much less than the pre-load spring force and contact is maintained between the fuel clad and the grid spring and dimples. Fatigue of the clad and fretting between the clad and grid support are not anticipated.

The effect of thermal cycling on the grid-clad support is merely a slight relative movement between the grid contact surfaces and the clad, which is gradual in nature during heat-up and cool-down. Since the number of cycles of the occurrence is small over the life of a fuel assembly (~ 5 years), negligible wear of the mating parts is expected.

The dynamic deflection of the control rods and the burnable poison rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot. With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed cyclic deflection through the available clearance gap results in an insignificantly low stress in either the clad tubing or in the flexure joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider, or the retainer plate in the case of the burnable poison rods.

A calculation, assuming the rods are supported by the surface of the dashpots and at the upper end by the spider or retainer, results in a similar conclusion.

3.2.3.4 RCCA Drive Mechanism

3.2.3.4.1 Design Description

The magnetic latch RCCA drive mechanisms are used for withdrawal and insertion of the rod cluster control assemblies into the reactor core and to provide sufficient holding power for stationary support.

Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity.

The complete drive mechanism, shown in Figure 3.2-14, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the position indicator coil stack.

Each assembly is an independent unit, which can be dismantled or assembled separately. The reactor vessel head and CRDMs utilize butt welds to attach the rod travel housing to the control rod drive mechanism. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the RCCA.

Reactor coolant fills the pressure containing parts of the drive mechanism. All working components and the shaft are immersed in the coolant.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing, induce magnetic flux through the housing wall to operate the working components. They move two sets of latches, which lift or lower the grooved drive shaft.

The three magnets are turned on and off in a fixed sequence by solid-state switches for the rod assemblies.

The sequencing of the magnets produces step motion over the 144 inches of normal control rod travel.

The mechanism develops a minimum lifting force of three times the static load. Therefore, extra life capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adapter will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

A multi-conductor cable connects the mechanism operating coils to the 125V dc power supply. The power supply is described in Chapter 7.

3.2.3.4.2 Latch Assembly

The latch assembly contains the working components, which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets. They actuate two sets of latches, which engage the grooved section of the drive shaft.

The upper set of latches moves up or down to raise or lower the drive rod by 5/8 inch. The lower set of latches has a maximum 1/16-inch axial movement to shift the weight of the control rod from one set of latches to the other. In the de-energized condition, the latch assembly does not engage the drive shaft.

3.2.3.4.3 Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

The housings are designed in accordance with the requirements for Class A vessels of the ASME Nuclear Vessel Code.

3.2.3.4.4 Operating Coil Stack

The operating coil stack is an independent unit, which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and can be removed or installed while the reactor is pressurized.

The operator coils (A, B, and C) are made of round copper wire which is insulated with a double layer of filament type glass yarn.

Coil temperatures are not monitored during operation of the plant.

The design operating temperatures can only be exceeded if the airflow to the mechanisms decreases to a value lower than the minimum requirements. If for some reason the flow is lower than the requirement, the air temperature monitoring devices will indicate need to actuate the standby fan to provide the minimum required flow. Only when both fans cannot supply the minimum flow will the plant be shut down until corrective measures restore the minimum air flow requirements.

In the unlikely event that the design operating temperatures of the control rod drive mechanism coils were to be exceeded, over a period of time, the life of the coil would be degraded and hence the plant would be shut down to avoid coil damage.

Failure of one coil would at most result in the dropping of a single rod into the core. Operation with one rod in the core is permitted by Technical Specification 3.10. Failure of a CRDM coil will not affect position-determining capability.

3.2.3.4.5 Drive Shaft Assembly

The main function of the drive shaft is to connect the RCCA to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 144 inches of RCCA travel. The grooves are spaced 5/8 inch apart to coincide with the mechanism step length and have 45 degree-angle sides.

The drive shaft is attached to the RCCA by the coupling. The coupling has two flexible arms, which engage the grooves in the spider assembly.

A ¼-inch diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod. At its upper end, there is a disconnect assembly for remote disconnection of the drive shaft assembly from the RCCA. During plant operation, the drive shaft assembly remains connected to the RCCA at all times.

3.2.3.4.6 Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing. It detects drive rod position by means of cylindrically wound differential transformers, which span the normal length of RCCA travel (144 inches).

3.2.3.4.7 Drive Mechanism Materials

All parts exposed to reactor coolant, such as the pressure vessel, latch assembly and drive rod, are made of metals, which resist the corrosive action of the water.

Three types of metals are used exclusively: stainless steels, Inconel X, and cobalt-based alloys. Wherever, magnetic flux is carried by pressure-containing parts exposed to the reactor coolant stainless steel is used. Cobalt-based alloys are used for the pins and latch tips.

Inconel X is used for the springs of both latch assemblies and 304 stainless steel is used in all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts (such as threads) during assembly.

Outside of the pressure vessel, where the metals are exposed only to the reactor plant containment environment and cannot contaminate the reactor coolant, carbon and stainless steels are used. Carbon steel, because of its high permeability, is used for flux return paths around the operating coils. It is zinc-plated 0.001-inch thick to prevent corrosion.

3.2.3.4.8 Principles of Operation

The drive mechanisms shown schematically in Figure 3.2-15, withdraw and insert their respective control rods as electrical pulses are received by the operator coils.

ON and OFF sequence, repeated by solid-state switches in the power programmer, causes either withdrawal or insertion of the RCCA. Position of the RCCA is indicated by the differential transformer action of the position indicator coil stack surrounding the rod travel housing. The differential transformer output changes as the top of the ferromagnetic drive shaft assembly moves up the rod travel housing.

Generally, during plant operation, the drive mechanisms hold the RCCAs withdrawn from the core in a static position, and only the stationary gripper coil is energized on each mechanism.

RCCA Withdrawal: The RCCA is withdrawn by repeating the following sequence:

1. Movable Gripper Coil ON

   The movable gripper armature rises and swings the movable gripper latches into the drive shaft groove.

2. Stationary Gripper Coil OFF

   Gravity causes the stationary gripper latches and armature to move downward until the load of the drive shaft is transferred to the movable gripper latches. Simultaneously, the stationary gripper latches swing out of the shaft groove.

3. Lift Coil ON

   The gap between the lift armature and the lift magnet pole closes and the drive rod rises one step length (5/8 inch).

4. Stationary Gripper Coil ON

   The stationary gripper armature rises and closes the gap below the stationary gripper armature, and swings the stationary gripper latches into a drive shaft groove. The latches contact the shaft and lift it a maximum 1/16 inch. The drive rod load is so transferred from the movable to the stationary gripper latches.

5. Movable Gripper Coil OFF

   The movable gripper armature separates from the lift armature under the force of a spring and gravity. Three links pinned to the movable gripper armature swing the three movable gripper latches out the groove.

6. Lift Coil OFF

   The gap between the lift armature and the lift magnet pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to the next groove.

RCCA Insertion: The sequence for RCCA insertion is similar to that for control rod withdrawal:

1. Lift Coil - ON

   The movable gripper latches are raised to a position adjacent to a shaft groove.

2. Movable Gripper Coil - ON

   The movable gripper armature rises and swings the movable gripper latches into a groove.

3. Stationary Gripper Coil - OFF

   The stationary gripper armature moves downward and swings the stationary gripper latches out of the groove.

4. Lift Coil - OFF

   Gravity separates the lift armature from the lift magnet pole and the control rod drops down 5/8 inch.

5. Stationary Gripper Coil - ON

6. Movable Gripper Coil - OFF

The sequences described above are termed as one step or one cycle and the control rod moves 5/8 inch for each cycle. Each sequence can be repeated at a rate of up to 72 steps per minute and the RCCAs can therefore be withdrawn or inserted at a rate of up to 45 inches per minute.

3.2.3.4.9 RCCA Tripping

If power to the stationary gripper coil is cut off, as for tripping, the combined weight of the drive shaft and the RCCA is sufficient to move the latches out of the shaft groove. The RCCA falls by gravity into the core. The tripping occurs as the magnetic field, holding the movable gripper armature against the lift magnet, collapses and the movable gripper armature is forced down by the weight acting upon the latches.

3.2.3.4.10 Part-Length Rods Drive Mechanisms

The part-length rods have been removed from the core. The drive mechanisms remained installed in the reactor vessel head until the reactor vessel head was replaced in 2004.

3.2.3.4.11 CRDM Housing Mechanical Failure Evaluation

An evaluation of the possibility of damage to adjacent control rod drive mechanism housings in the event of a circumferential or longitudinal failure of a rod housing located on the vessel head is presented.

An RCCA drive mechanism schematic is shown in Figure 3.2-15. The operating coil stack assembly of this mechanism has a 10.8-inch by 10.8-inch cross section and a 39.875-inch length. The position indicator coil stack assembly (not shown in this figure) is located above the operating coil stack assembly. It surrounds the rod travel housing over nearly its entire length. The rod travel housing outside diameter is 3.8 inches and the position indicator coilstack assembly consists of an 1/8-inch thick stainless steel tube surrounded by a continuous stack of copper wire coils. This assembly is held together by two end plates (the top end plate is square), an outer sleeve, and four axial tie rods.

3.2.3.4.12 Effect of Rod Travel Housing Longitudinal Failures

Should a longitudinal failure of the rod travel housing occur, the region of the stainless steel tube opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the stainless steel tube. A radial free-water jet is not expected to occur because of the small clearance between the stainless steel tube and the rod travel housing, and the considerable resistance of the combination of the stainless steel tube and the position indicator coils to internal pressure. Calculations based on the mechanical properties of stainless steel and copper at reactor operating temperature show that an internal pressure of at least 4000 psia would be necessary for the combination of the stainless steel tube and the coils to rupture.

Therefore, the combination of stainless steel tube and copper coils stack is more than adequate to prevent formation of a radial jet following a control rod-housing split, which assures the integrity of the adjacent rod housings.

3.2.3.4.13 Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil stack assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited to less than 2 feet by the missile shield, thereby limiting the projectile acceleration. When the projectile reaches the missile shield, it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would push the broken-off piece against the missile shield.

If the broken-off piece were short enough to clear the break when fully ejected, it could rebound after impact with the missile shield. The top end plates of the position indicator coil stack assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage.

3.2.3.4.14 Summary

The considerations given above lead to the conclusion that failure of a control rod housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housing that would increase the severity of the initial accident.

3.2 References

1. Moore, J. S., “Nuclear Design of Westinghouse Pressurized Water Reactor With Burnable Poison Rods,” WCAP-7806, December 1971
2. Moore, J. S., “Power Distribution Control in Westinghouse Pressurized Water Reactors,” WCAP-7811, December 1971
3. Brandon, R. F., R. F. Barry, J. F. Kutz, et al., “Power Distribution Monitoring in the R. E. Ginna PWR,” WCAP-7756, September 1970
4. Wood, P. M., E. C. Bassler, et al., “Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors,” WCAP-7113, October 1967
5. “Reactor Test Program, Kewaunee Power Station,” (Revision 9, September 2006)
6. PD2-01-46, Rev. 1 (01WP-S-002), M. L. Lewis, M. A. Marzean and A. Lahlou, “Kewaunee Fuel Transition Work Report, Revision 1 to Fuel Assembly Compatibility Report for the Supply of 14x14 Westinghouse 422V+ Fuel Assembly,” November 1, 2001
7. Friedland, A. J. and S. Ray, “Revised Thermal Design Procedure,” WCAP-11397-P-A, April 1989
8. Motley, F. E., K. W. Hill, F. F. Cadek, J. Shefcheck, “New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids,” WCAP-8762-P-A, July 1984
9. Letter from M. F. Baumann, (WEPCO) to Document Control Desk (NRC), “14x14, 0.422" OD VANTAGE + (422V+) Fuel Design,” NPL 97-0538, November 1997

10. Sung, Y. X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary), October 1999.

11. Moomau, W. H., "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis and 54F Replacement Steam Generators)," WCAP-15591, Rev. 0, May 2002

12. Spier, E. M., "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7308-L-P-A, June 1988

13. Skaritka, J., "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1, July 1979

14. Letter from E. P. Rahe, Jr. (Westinghouse) to J. R. Miller, (NRC), "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NS-EPR-2515, October 9, 1981; and Letter from E. P. Rahe, Jr. (Westinghouse) to J. R. Miller, (NRC), "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NS-EPR-2572, March 16, 1982

15. Letter from C. Berlinger, (NRC) to E. P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986

16. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

17. Schueren, P. and McAtee, K. R., "Extension of Methodology for Calculating Transition Core DNBR Penalties," WCAP-11837, May 1988.

18. Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-11837," October 1989

19. "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation Report EMF-92-153 (P)(A) and Supplement 1, March 1994

20. Laufer, R. J., (NRC) to M. L. Marchi (WPS), "Safety Evaluation for the High-Thermal Performance Departure from Nucleate Boiling Correlation (DNB) and the Associated 1.14 Minimum DNB Ratio Safety Limit for Kewaunee Fuel," Letter No. K-98-001, December 30, 1997

21. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report WPSRSEM-NP-A entitled, "Reload Safety Evaluation Methods for Application to Kewaunee," Revision 3, October 2001

22. WCAP-7113, "Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors," October 1967

23. Davis, M. J., (NRC) to K. H. Evers (WPS), transmitting the NRC SER for Amendment No. 92 to the Operating License, approving the increase in allowable fuel enrichment, Letter No. K-91-048, March 7, 1991

24. EMF-2622(P), "Mechanical Design Evaluation for Kewaunee Reload Kew-21 and Cycle 25 Fuel Assemblies," Framatome ANP, July 2001

25. "Kewaunee LUA Fuel Rod Design Work Report - Fuel Performance Analysis Report for the Supply of 14x14 Westinghouse 422V+ Fuel Assemblies," by A. Meliksetian, August 28, 2001

26. WCAP-15425, "Determination of Acceptable Baffle-Barrel-Bolting for the Kewaunee and Prairie Island Plants," May 2001

Table 3.2-1
Nuclear Design Data

Structural Characteristics	
Fuel Weight (UO_2) (lb), Typical Core (FRA-ANP Standard/FRA-ANP Heavy/Westinghouse 422V+)	116,000/123,000/122,000
Zircaloy Weight (lb), Typical Core (FRA-ANP Standard/FRA-ANP Heavy/Westinghouse 422V+)	29,000/24,500/23,800
Core Diameter (in.)	96.5
Core Height (in.) (FRA-ANP Heavy/FRA-ANP Standard/Westinghouse 422V+)	144/144/143.25
Reflector Top-Water Plus Steel (in.)	~10
Reflector Bottom-Water Plus Steel (in.)	~10
Reflector Side-Water Plus Steel (in.)	~15
H_2O/U Volume Ratio (Cold) (FRA-ANP Standard/FRA-ANP Heavy/Westinghouse 422V+)	3.85/3.65/3.70
Number of Fuel Assemblies	121
Number of UO_2 Rods per Assembly	179
Performance Characteristics	
Total Reactor Heat Output, (MWt)	1772[1]
Fuel Burnup, (MWD/MTU), Typical Cycle	17, 500
Fuel Enrichment (weight %) Typical Cycle	4.5 to 4.9
Nuclear Heat Flux Hot Channel Factor, F^N_q Fullpower (FRA-ANP Standard/FRA-ANP Heavy/Westinghouse 422V+)	As specified in the COLR
Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$ Fullpower (FRA-ANP Standard/FRA-ANP Heavy/Westinghouse 422V+)	As specified in the COLR
Control Characteristics	
Rod Cluster Control Assemblies (RCCAs) Material	5% Cd; 15% In; 80% Ag
Number of RCCAs (Full Length)	29
Number of RCCAs (Part Length)	0
Number of Absorber Rods per RCCA Assembly	16
Total RCCA Worth, (Typical Cycle)	(See Table 3.2-2)

Table 3.2-1
Nuclear Design Data

Control Characteristics (con't)

Discreate Burnable Poison Rods Material		Borosilicate Glass
Boron 10 Loading (gm/in)		0.01885
Integral Burnable Poison Material		Gadolinia
Gadolinia Weight Percent		2, 4, 6 or 8
Soluble Boron Concentrations		
Refueling Shutdown -	Rods in ($k \leq 0.95$) (ppm) Rods out ($k \leq 0.99$) (ppm)	As specified in the COLR
Hot Zero Power Beginning of Cycle No Xenon (Typical) (ppm)		1980
Hot Fullpower Beginning of Cycle No Xenon (Typical) (ppm)		1750
Hot Fullpower Beginning of Cycle EQ Xenon (Typical) (ppm)		1400
Reactivity Characteristics Design Range		
Moderator Temperature Coefficient (PCM/°F_m)		+5.0[2] to -40.0
Moderator Pressure Coefficient (pcm/psi)		-.03 to +.35
Moderator Density Coefficient, cm^3/gm		-0.10 to +0.30
Doppler Coefficient (pcm/°F_f)		-1.0 to -2.32
Delayed Neutron Fraction,%		0.485 to 0.706
Prompt Neutron Lifetime, sec		1.45E-5 to 3.0E-5
Differential Boron Worth (pcm/ppm)		-6.0 to -11.2

1. The Reactor was originally licensed for a reactor heat output of 1650 MWt. In 2004, a license amendment was received that increased the reactor heat output to 1772 MWt. The basis of the change was a 1.4% measurement uncertainty recapture power uprate and a 6% stretch uprate described in Chapter 14.

2. Positive moderator coefficient is allowed only at power levels less than 60%.

Table 3.2-2
Shutdown Margin Analysis (Typical Core Design)

	BOC (pcm)	EOC (pcm)
RCCA Worth – All Rods Inserted	5500	6280
Most Reactive Stuck Rod	580	660
Net	4920	5620
8.5% Margin	420	480
Net Rod Worth	4500	5140
Total Reactivity Requirements (includes Doppler, Moderator Temperature, Voiding, Flux Redistribution and Rod Insertion Allowance)	2070	3690
Calculated Shutdown Margin (Net Rod Worth-Total Reactivity Requirements)	2430	1450
Required Shutdown Margin (The required SDM is documented in the cycle specific COLR)	1430	1430

Table 3.2-3
Kewaunee Thermal And Hydraulic Design Parameters

	Parameter Value
Thermal-Hydraulic Design Parameters (using RTDP)	
Reactor Core Heat Output, MWt	1772
Reactor Core Heat Output, 10^6, BTU/Hr	6046
Heat Generated in Fuel,%	97.4
Core Pressure, Nominal – RTDP, psia	2265
Pressurizer Pressure, Nominal, psia	2250
Radial Power Distribution Limits	As specified in the COLR
HFP Nominal Coolant Conditions	
Vessel Thermal Design Flow (TDF) Rate (including bypass)	
10^6 lbm/hr	67.87
GPM	178,000
Core Flow Rate (excluding Bypass,[2] based TDF)	
10^6 lbm/hr	63.11
GPM	165,540
Core Flow Area, ft^2	27.1 (full-core 422V+)
Core Inlet Mass Velocity, (based on TDF)	
10^6 lbm/hr-ft^2	2.33
Thermal-Hydraulic Design Parameters (based on TDF)	
Nominal Vessel/Core Inlet Temperature, °F	539.2
Vessel Average Temperature, °F	573.0
Core Average Temperature, °F	577.1
Vessel Outlet Temperature, °F	606.8
Core Outlet Temperature, °F	611.3
Average Temperature Rise in Vessel, °F	67.6
Average Temperature Rise in Core, °F	72.1

Table 3.2-3
Kewaunee Thermal And Hydraulic Design Parameters

	Parameter Value
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	28,507
Average Heat Flux, BTU/hr-ft^2	206,600
Average Linear Power, kW/ft	6.87
Peak Linear Power for Normal Operation,(3) kW/ft	17.18
Peak Linear Power for Prevention of Centerline Melt, kW/ft	22.54
Mechanical Design Flow, gpm per Loop	102,800
Minimum Total Measured Flow, gpm	186,000
Pressure Drop Across Core, psi(4)	22.7

Notes:
2. Based on design bypass flow of 7%.

3. Based on maximum F_Q of 2.50.

4. Based on best estimate reactor flow rate of 98,900 gpm/loop.

Table 3.2-4
Peaking Factor Uncertainties

$$F_{\Delta H} = F^N_{\Delta H} \text{ x } F^E_{\Delta H}$$

where:	$F^N_{\Delta H}$	Nuclear Enthalpy Rise Hot Channel Factor – The ratio of the relative power of the hot rod, which is one of the rods in the hot channel, to the average rod power. The normal operation value of this is given in the plant Technical Specifications or a Core Operating Limit Report (COLR).
	$F^E_{\Delta H}$	Engineering Enthalpy Rise Hot Channel Factor – The nominal enthalpy rise in an isolated hot channel can be calculated by dividing the nominal power into this channel by the core average inlet flow per channel. The engineering enthalpy rise hot channel factor accounts for the effects of flow conditions and fabrication tolerances. It can be written symbolically as:

$$F^E_{\Delta H} = f\,(F^E_{\Delta H,1}, F^E_{\Delta H,2}, F^E_{\Delta H \text{ inlet maldist}}, F^E_{\Delta H \text{ redist}}, F^E_{\Delta H \text{ mixing}})$$

where:	$F^E_{\Delta H,1}$	accounts for rod-to-rod variations in fuel enrichment and weight
	$F^E_{\Delta H,2}$	accounts for variations in fuel rod outer diameter, rod pitch, and bowing
	$F^E_{\Delta H \text{ inlet maldist}}$	accounts for the nonuniform flow distribution at the core inlet
	$F^E_{\Delta H \text{ redist}}$	accounts for flow redistribution between adjacent channels due to the different thermal-hydraulic conditions between channels
	$F^E_{\Delta H \text{ mixing}}$	accounts for thermal diffusion energy exchange between adjacent channels caused by both natural turbulence and forced turbulence due to the mixing vane grids

The value of these factors and the way in which they are combined depends upon the design methodology used, that is, STDP or RTDP. Note that no actual combined effect value is calculated for $F^E_{\Delta H}$. These factors are accounted for by using the VIPRE-W code.

Table 3.2-5
RTDP Uncertainties

Parameter	Calculated Uncertainty	Uncertainty Used in Safety Analysis
Power	±1.72% power -0.32% power (bias)	±2.0% power 0.32% power (bias) (at 1757 MWt-NSSS power)
Reactor Coolant System Flow	±2.86% flow -0.11% flow (bias)	±4.3% flow -0.11% flow (bias)
Pressure	±35.1 psi 15.0 psi (bias)	±50.1 psi 15.0 psi (bias)
Inlet Temperature	±4.9°F -1.1°F	±6.0°F -1.1°F

Table 3.2-6
DNBR Margin Summary[1]

		422V+
DNB Correlation		WRB-1
DNBR Correlation Limit		1.17
DNBR Design Limit	(TYP)[2]	1.25
	(THM)[3]	1.25
DNBR Safety Limit	(TYP)	1.34
	(THM)	1.34
DNBR Retained Margin[4]	(TYP)	6.7%
	(THM)	6.7%
Rod Bow DNBR Penalty		2.6%
Transition Core DNBR Penalty		≤ 3.0%
Available DNBR Margin[5]	(TYP)	1.1%
	(THM)	1.1%

Notes:

1. Steam line break is analyzed using the W-3 correlation with STDP. The correlation limit DNBR is 1.45 in the range of 500 to 1000 psia. Rod withdrawal from subcritical is also analyzed using the W-3 correlation (w/o spacer factor) with STDP below the bottom NMV grid. The correlation limit DNBR is 1.30 above 1000 psia and the safety limit DNBR is 1.39 which provides 6.7% margin to cover the rod bow penalty and retain generic margin for operational issues. WRB-1 with RTDP is used for rod withdrawal from subcritical above the bottom NMV grid.
2. TYP ≡ Typical Cell
3. THM ≡ Thimble Cell
4. DNBR margin is the margin that exists between the safety limit and the design limit DNBRs.
5. The transition core analysis was performed at uprated conditions (that is, 1772 MWt). Additional margin will exist (such as, on the order of 12%) prior to plant uprating.

Table 3.2-7
Limiting Parameter Direction

Parameter	Limiting Direction for DNB
$F^N_{\Delta H}$, nuclear enthalpy rise hot-channel factor	maximum
Heat generated in fuel (%)	maximum
Reactor core heat output (MWt)	maximum
Average heat flux (BTU/hr-ft^2)	maximum
Nominal vessel/core inlet temperature (°F)	maximum
Core pressure (psia)	minimum
Pressurizer pressure (psia)	minimum
Thermal design flow for non-RTDP analyses (gpm)	minimum
Minimum measured flow for RTDP analyses (gpm)	minimum

Table 3.2-8
Core Mechanical Design Parameters [a]

Active Portion of the Core	
Equivalent Diameter, in.	96.5
Equivalent Diameter, in.	144
Active Fuel Height, in.	1.495
Length-to-Diameter Ratio	50.8
Total Cross-Section Area, ft^2	
Fuel Assemblies [d]	
Number	121
Rod Array	14 x14
Fuel Rods per Assembly	179 [b]
Rod Pitch, in.	0.556
Overall Dimensions, in.	7.763 x 7.763
Core-Fuel Weight (as UO_2), lb (FRA-ANP Standard/FRA-ANP Heavy/Westinghouse 422V+)	116,000/123,000/122,000
Core-Total Fuel Weight, lb (FRA-ANP Standard/FRA-ANP Heavy/Westinghouse 422V+)	152,800/155,200/153,200
Rod Cluster Control Assemblies	
Neutron Absorber	5% Cd, 15% In, 80% Ag
Cladding Material	Type 305 SS - Cold Worked
Clad Thickness, in.	0.019
Number of Clusters:	
Full Length	29
Part Length	4[(3)]
Number of Control Rods per Cluster	16
Weight in 60°F water:	114
Full-Length, lb	68 [c]
Part-Length, lb	158.454 (overall)
Length of Control Rods, in.	150.954 (insertion length)
Length of Absorber Section, in.	142.00 (full length) 36.00 (part length) [3]

Table 3.2-8

Core Mechanical Design Parameters [a]

Core Structure	
Core Barrel, in. ID / OD	109.0 / 112.5
Thermal Shield, in. ID / OD	115.3 / 122.5
Discrete Burnable Poison Rods (not used)	
Material	Borosilicate Glass
Outside Diameter, in.	0.431
Inner Tube, OD, in.	0.2365
Clad Material	S.S.
Inner Tube Material	S.S.
Boron Loading (Natural), gm/cm of Glass Rod	0.0429

a. All dimensions for cold conditions.

b. Seventeen rods are omitted; sixteen to provide passage for control rods and one to contain in-core instrumentation.

c. The part-length rod control system has been removed, although these rods were used in the initial core load.

d. See Table 3.2-9 for additional fuel assembly, rod and pellet information.

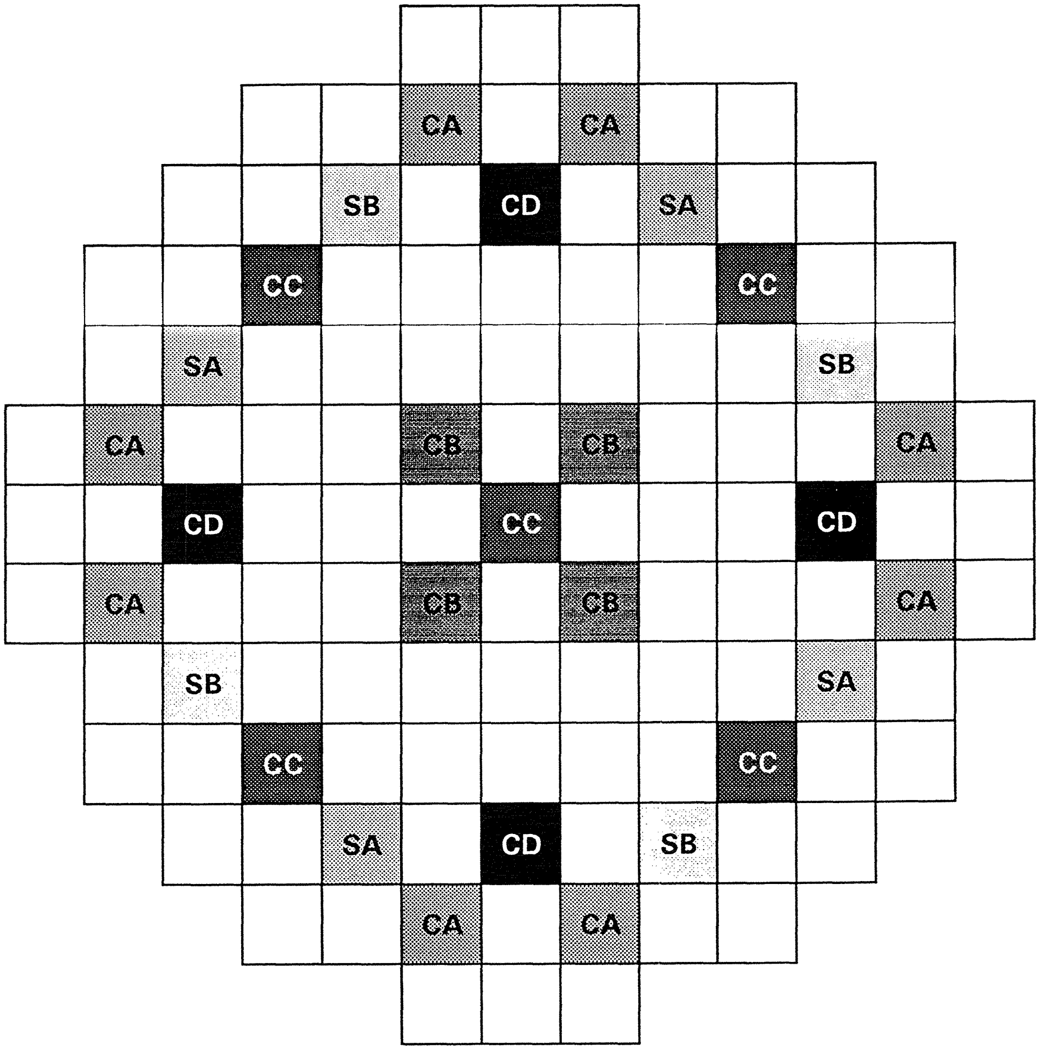
Table 3.2-9
Fuel Assembly and Component Descriptions

Component	Design Characteristics		
	FRA-ANP Standard Fuel	FRA-ANP Heavy Fuel	Westinghouse 422V+
Fuel Assembly			
Array	14x14	14x14	14x14
Pitch (Assy)	7.803 in.	7.803 in.	7.803 in.
Pitch (Rod)	0.556 in.	0.556 in.	0.556 in.
Length	159.7 in.	159.7 in.	159.8 in.
Distance between tie plates	153.6 in.	153.6 in.	153.9 in.
No. fuel rods	179	179	179
No. guide tubes	16	16	16
No. instrument tubes	1	1	1
No. spacers	7	7	7
Fuel (U) weight per assy (approx.)	384 kg	406 kg	403 kg
Total assy. weight (approx.)	1263 lb	1283 lb	1266 lb
Fuel Rods			
Total Length	152.07 in.	152.07 in.	152.56 in.
Active Fuel Length	144.0 in.	144.0 in.	143.25 in.
Upper Plenum Length	6.865 in.	6.865 in.	8.510 in.
Fuel/Clad Diametral Gap	0.0075 in.	0.0070 in.	0.0075 in.
Total Number in the core	21,659	21,659	21,659
Fuel Pellet			
Diameter	0.3565 in.	0.3670 in.	0.3659 in.
Length	0.428 in.	0.440 in.	0.439 in.
Theoretical Density	95.00%	95.35%	96.56%
Material	sintered UO_2	sintered UO_2	sintered UO_2
Cladding			
O.D.	0.424 in.	0.424 in.	0.422 in.
I.D.	0.364 in.	0.374 in.	0.3734 in.
Material	Zircaloy	Zircaloy	ZIRLO
Spacers (Top & Bottom/Mid)			
Height	1.55 in./2.25 in.	2.25 in./1.75 in.	1.904 in./2.672 in
Outer Dimension	7.763 in./7.763 in.	7.763 in./7.761 in.	7.759 in./7.756 in.
Material	Zr-4, Inconel 718/ Zr-4, Inconel 718	Zr-4, Inconel 718/ Zr-4	Inconel/ZIRLO

Table 3.2-9
Fuel Assembly and Component Descriptions

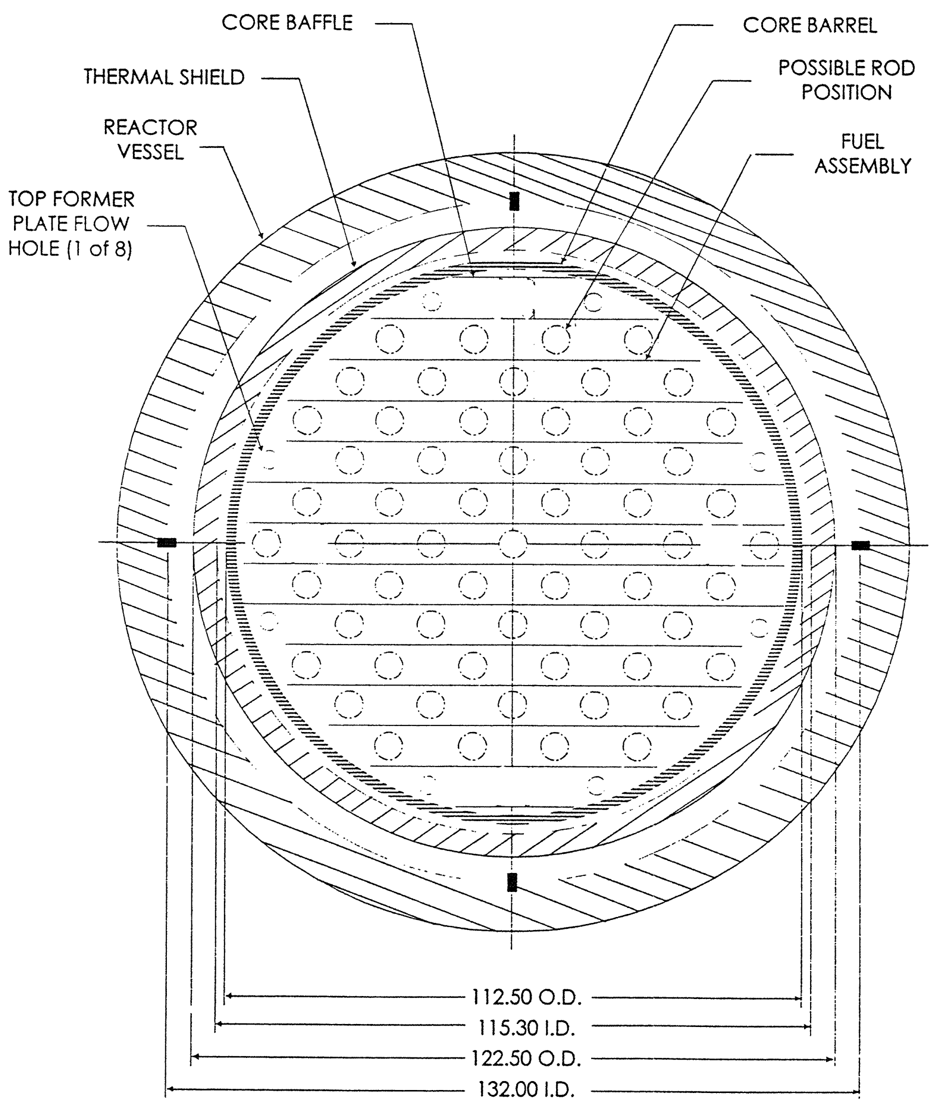
Component	Design Characteristics		
	FRA-ANP Standard Fuel	FRA-ANP Heavy Fuel	Westinghouse 422V+
Guide Tube			
Dashpot Length	24.62 in.	24.62 in.	24.074 in.
O.D. (above dashpot)	0.541 in.	0.541 in.	0.526 in.
I.D. (above dashpot)	0.507 in.	0.507 in.	0.492 in.
O.D. (below dashpot)	0.481 in.	0.481 in.	0.4815 in.
I.D. (below dashpot)	0.447 in.	0.447 in.	0.4465 in.
Overall length	154.0 in.	154.0 in.	152.7n in.
Material	Zr-4	Zr-4	ZIRLO

Figure 3.2-1
Rod Control Cluster Groups



Control Bank (A)	8
Control Bank (B)	4
Control Bank (C)	5
Control Bank (D)	4
Shutdown	8
TOTAL	29

Figure 3.2-2
Reactor Core Cross-Section



PLAN VIEW OF CORE - 121 FUEL ASSEMBLIES

Figure 3.2-3
Reactor Vessel Internals

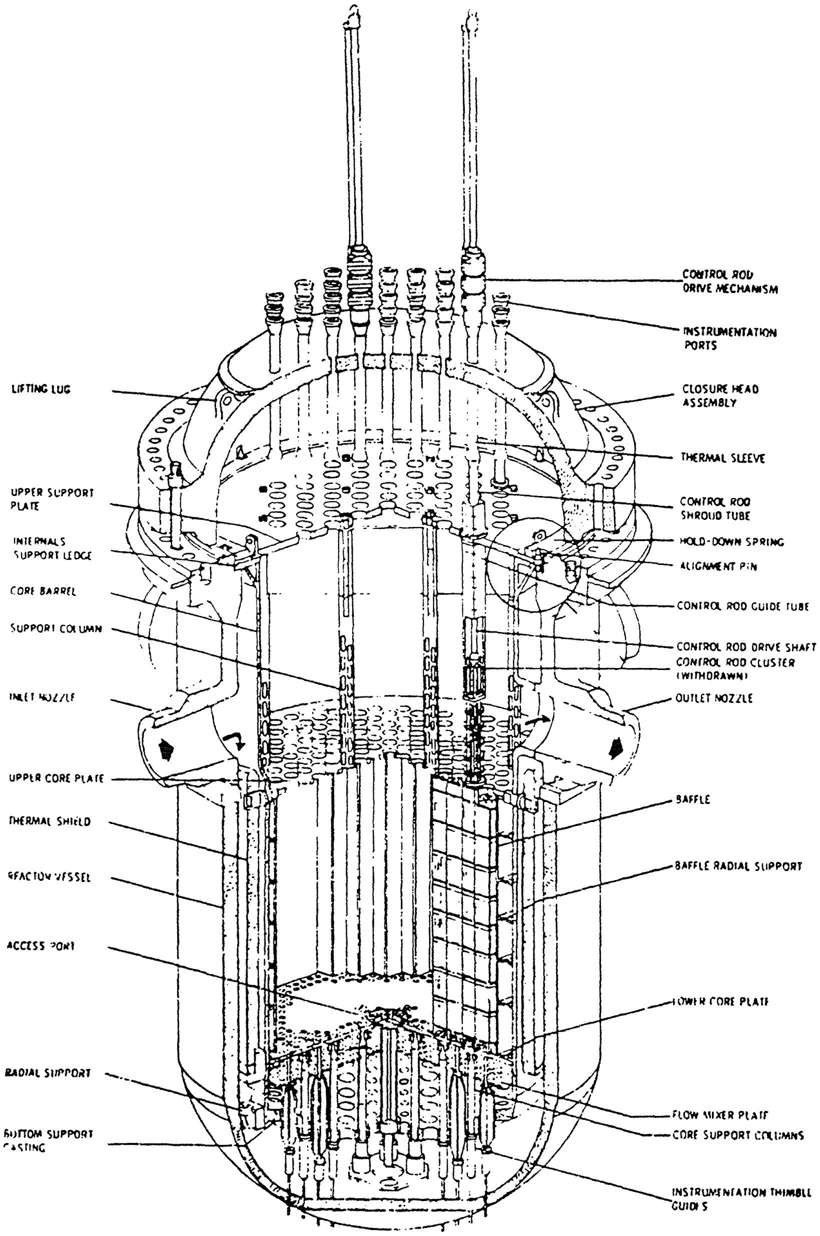
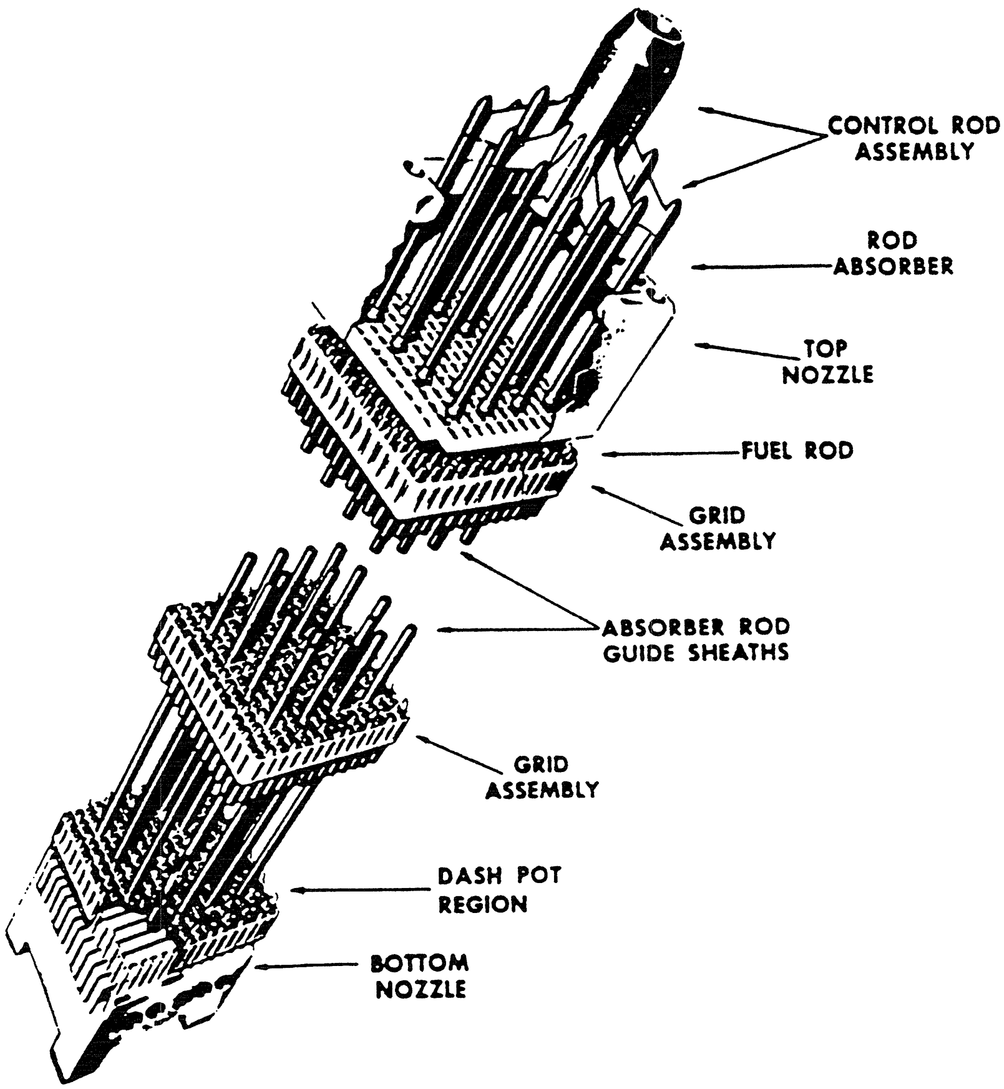


Figure 3.2-4
Typical Rod Cluster Control Assembly



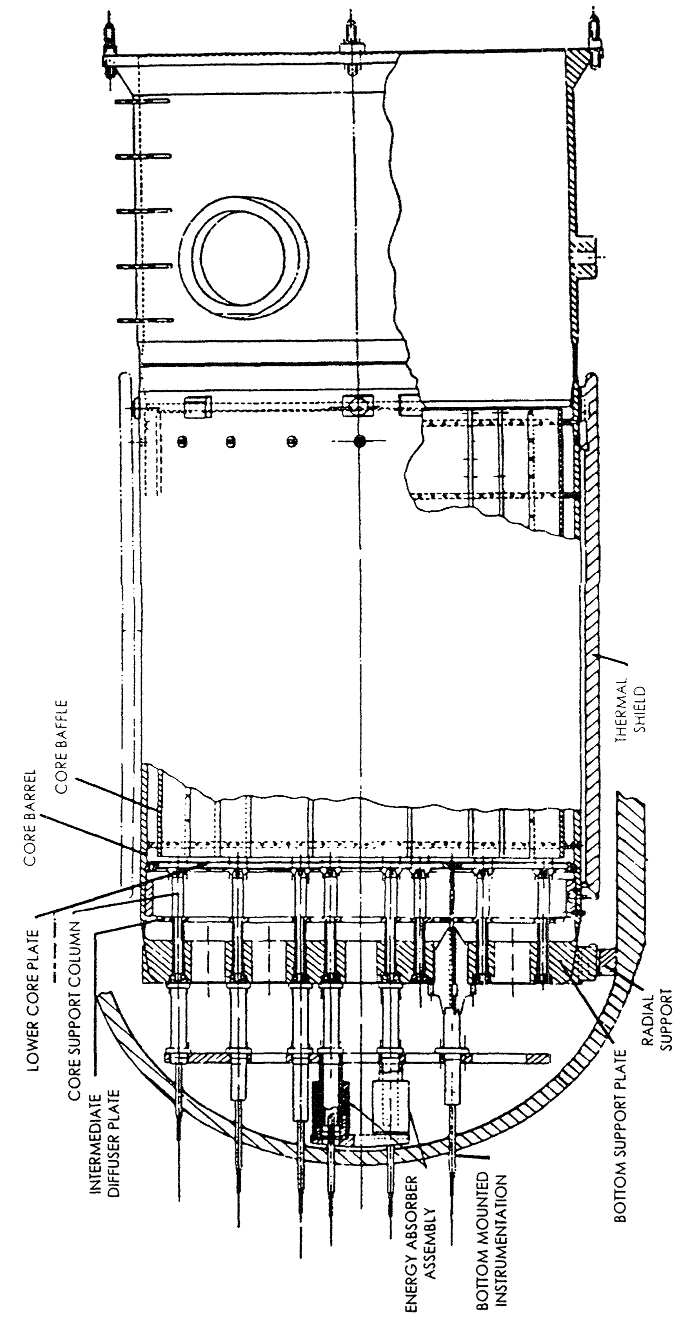


Figure 3.2-5
Lower Core Support Structure

Figure 3.2-6
Upper Core Support Assembly

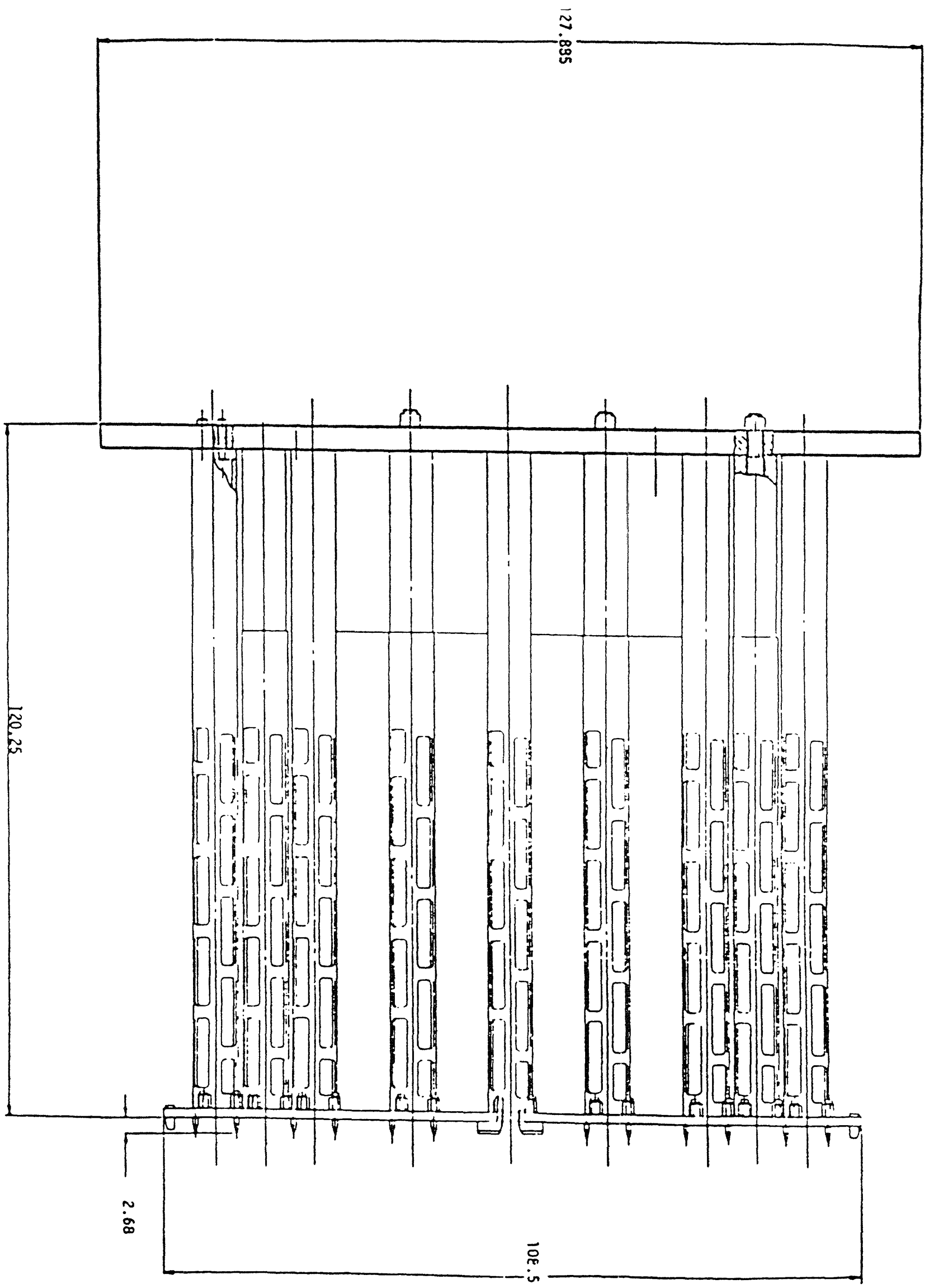


Figure 3.2-7
Guide Tube Assembly

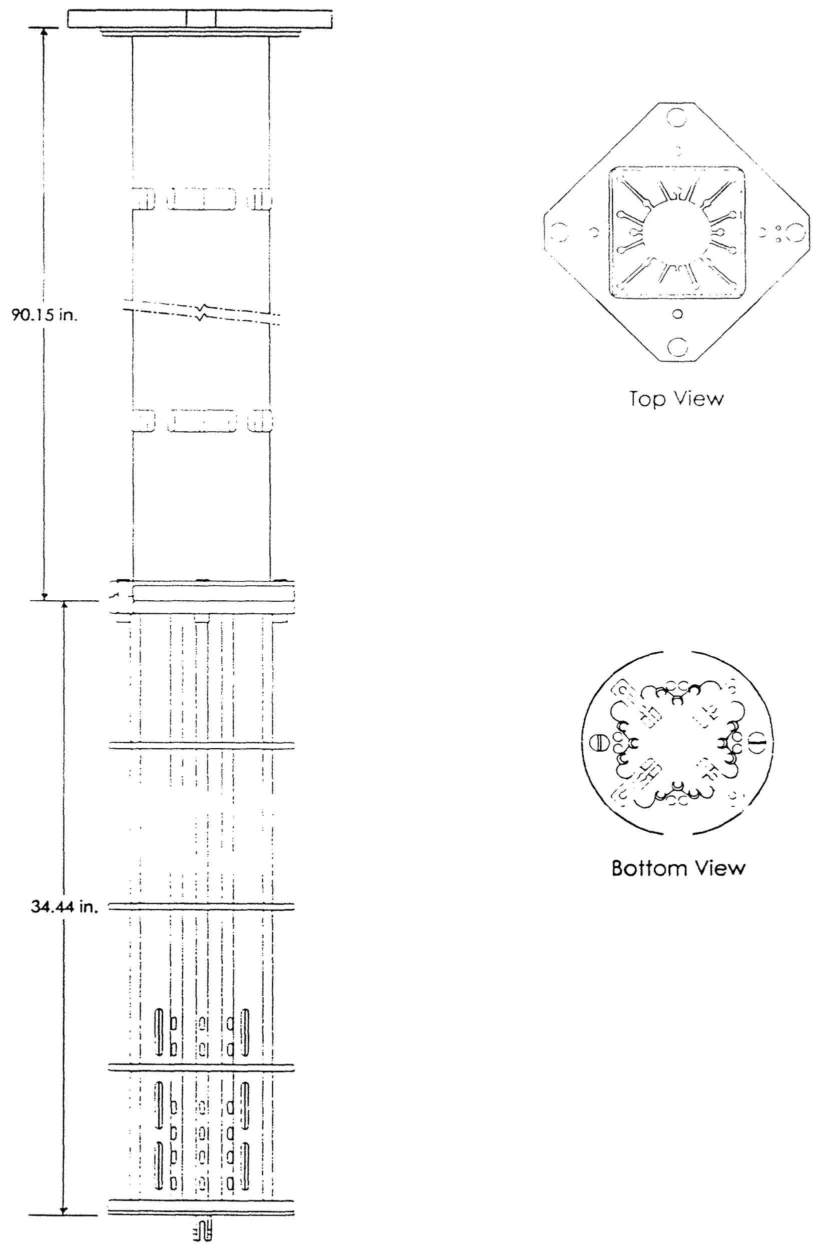
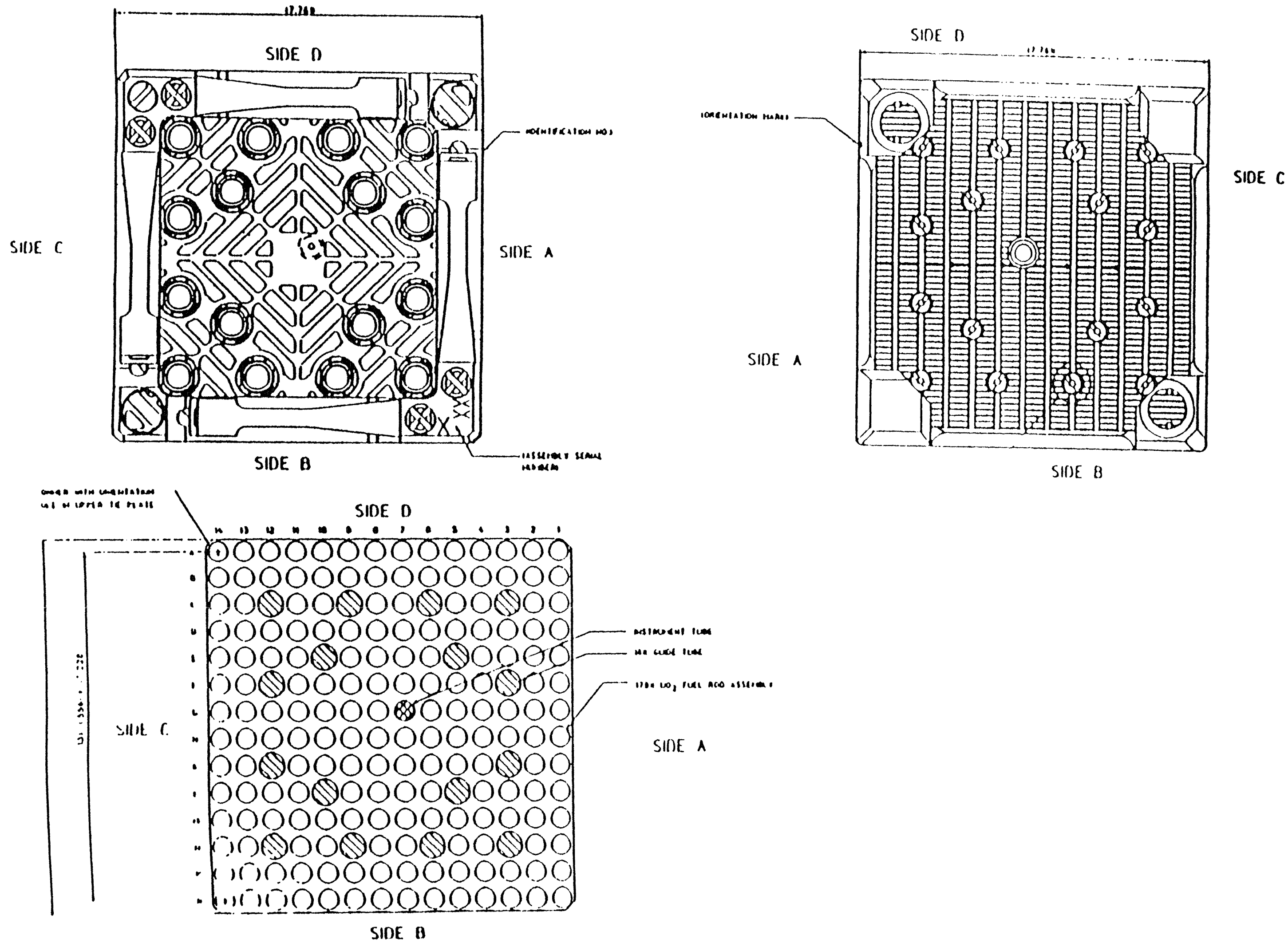


Figure 3.2-8
Framatome ANP (FRA-ANP) Fuel Assembly - Top View



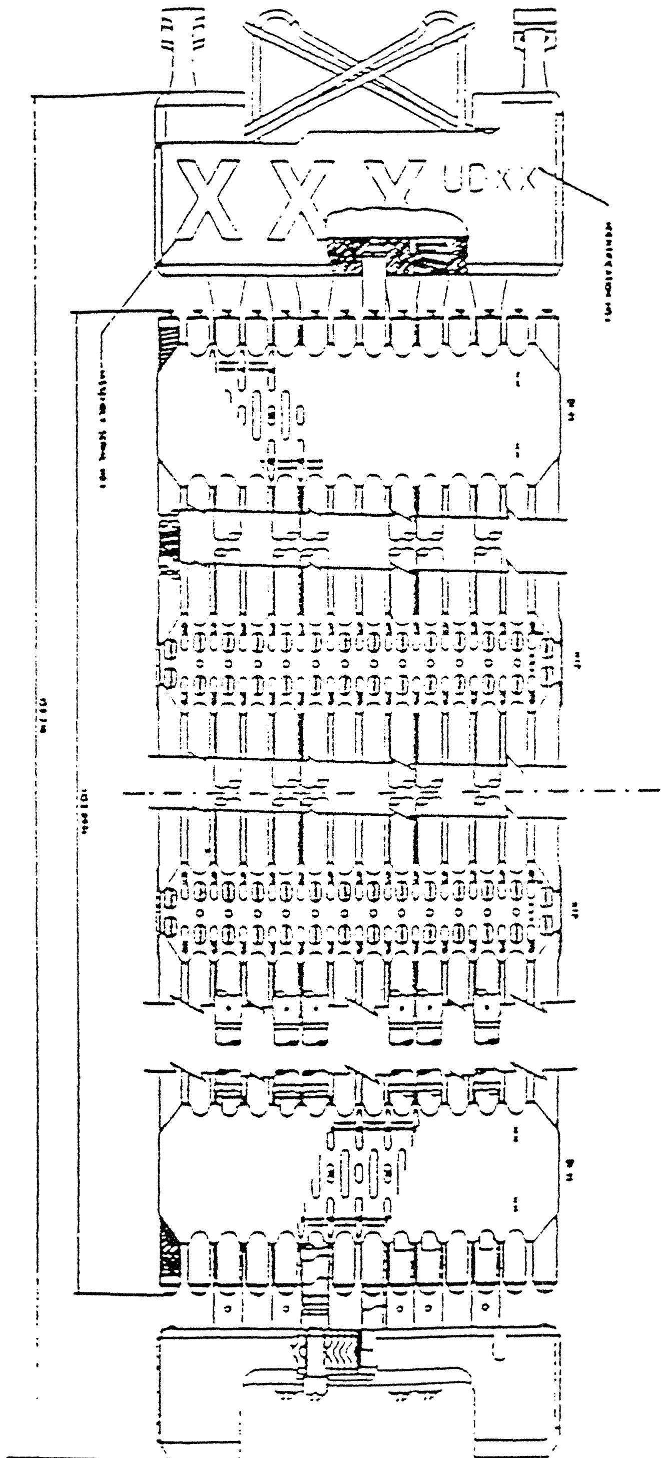


Figure 3.2-9
Framatome ANP (FRA-ANP) Fuel Assembly - Side View

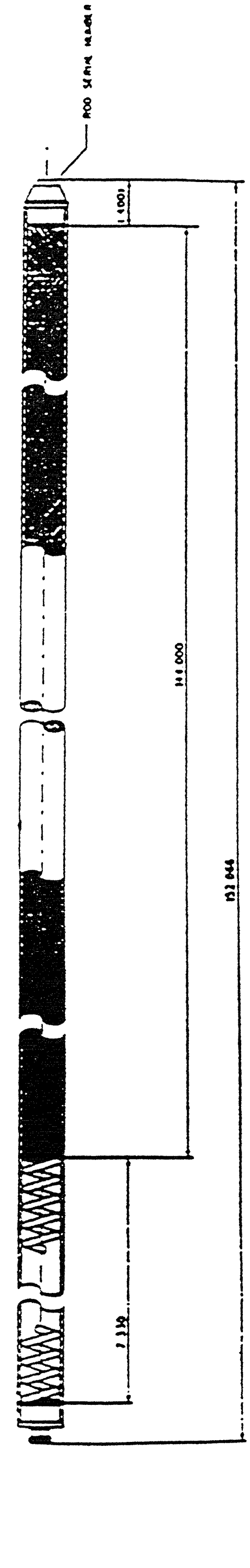


Figure 3.2-10
Framatome ANP (FRA-ANP) Fuel Rod

Figure 3.2-11
FUELGAURD™ Lower Tie Plate

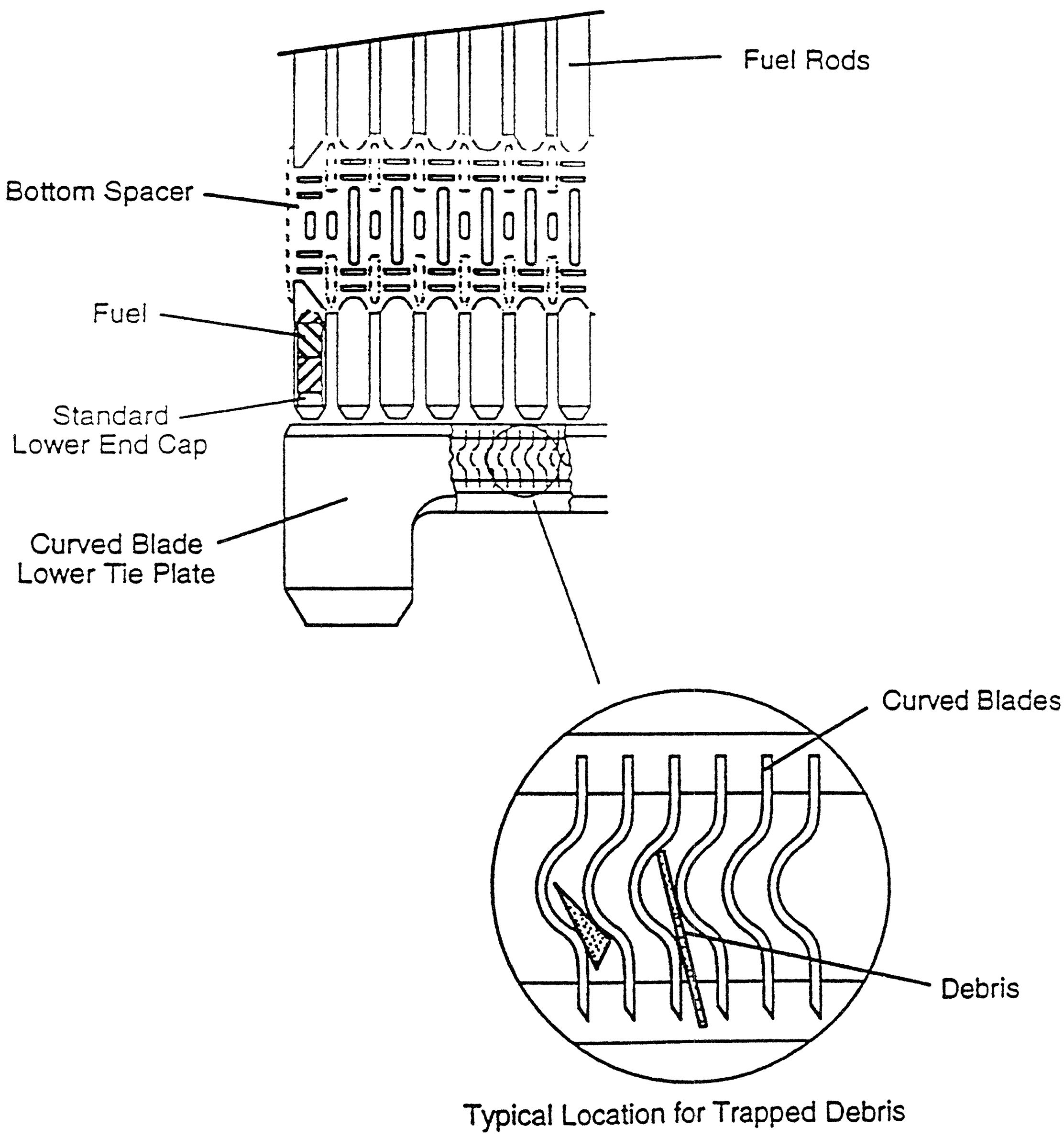


Figure 3.2-12
HTP Assembly

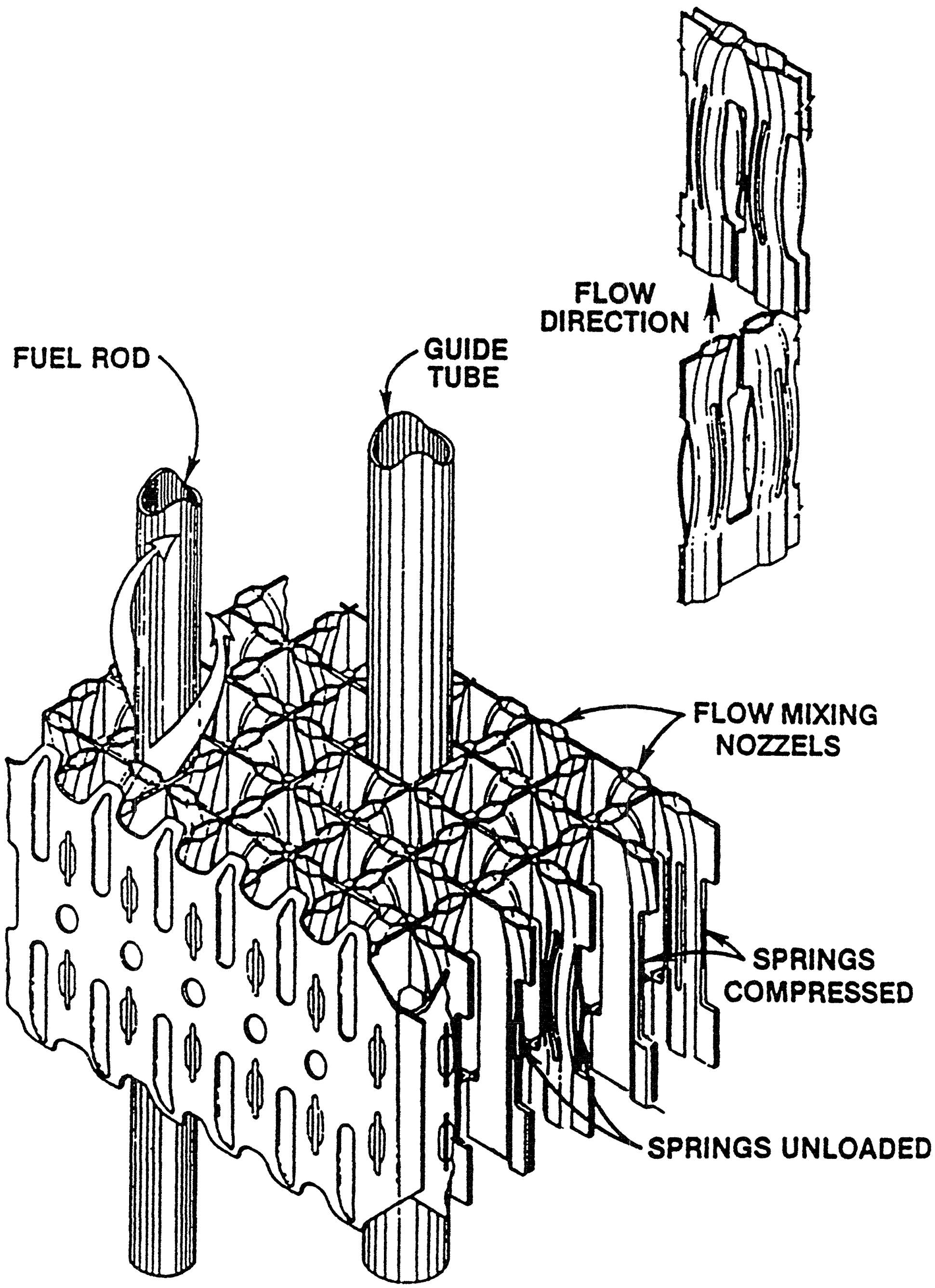


Figure 3.2-13
Detail of Standard Burnable Poison Rod

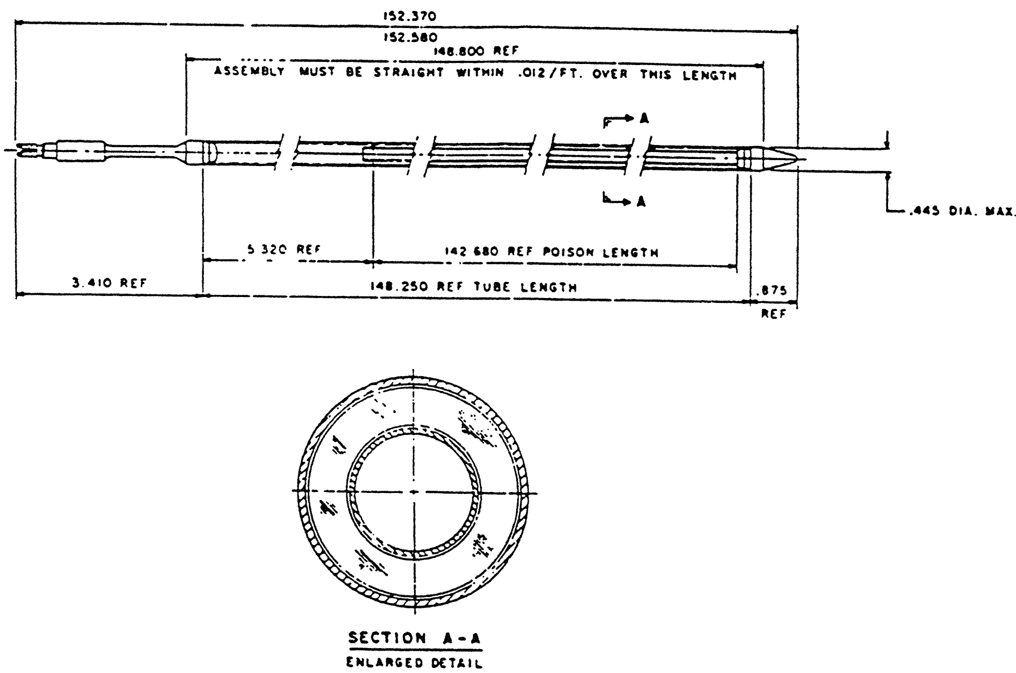


Figure 3.2-14
RCCA Drive Mechanism Assembly

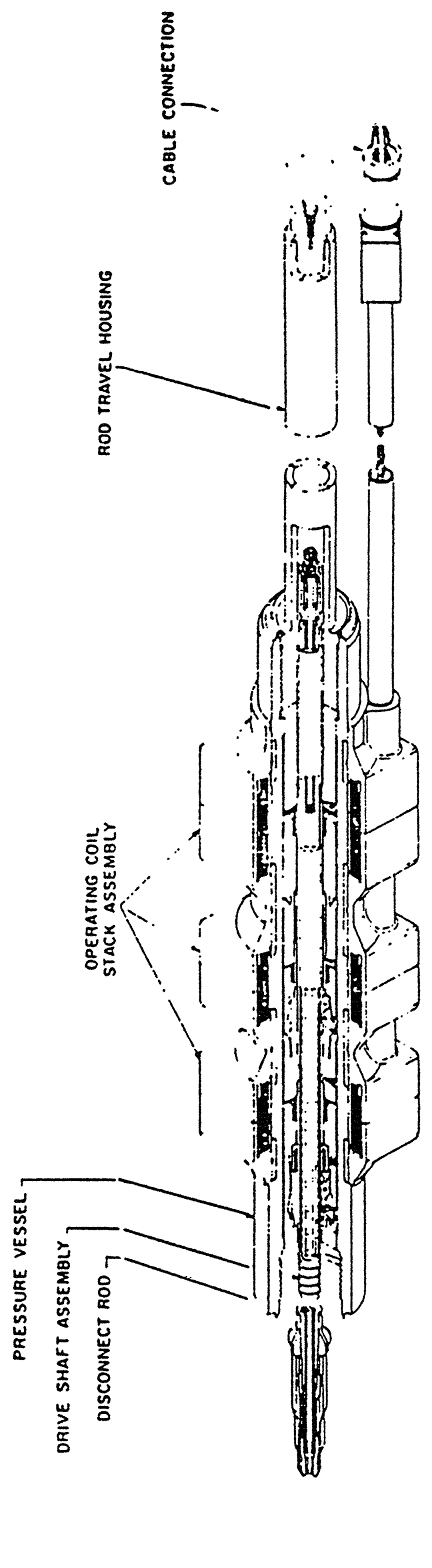
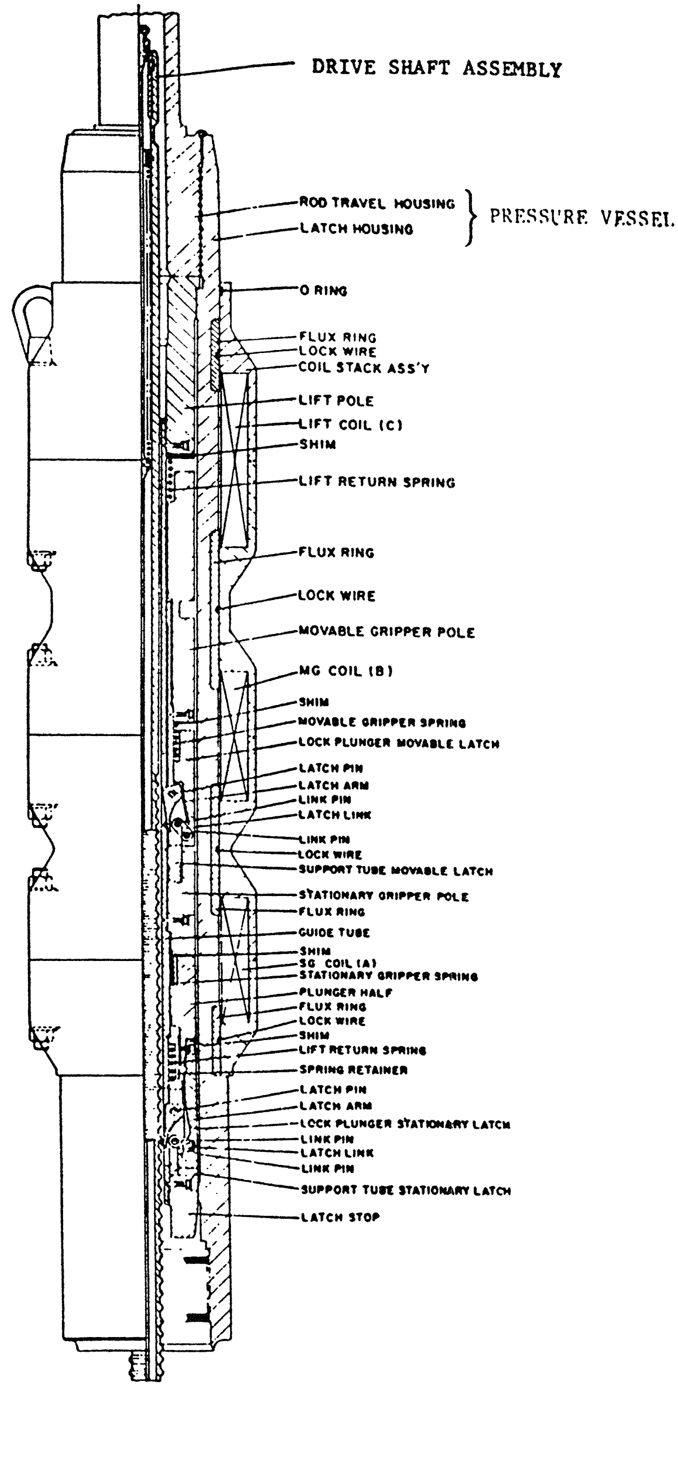


Figure 3.2-15
RCCA Drive Mechanism Schematic



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3.3 TESTS AND INSPECTIONS

3.3.1 Reactivity Anomalies

Reactor physics tests are performed each reload cycle (see Reference 1). Agreement between core design predictions and measurements, as specified by the reactor test program, verifies that the reactivity and power distribution of the actual core are acceptable and no anomalies are present. Surveillance procedures performed during plant operation provide continued assurance that there no reactivity or power distribution anomalies.

Nuclear flux channels are calibrated periodically when the plant is at power, using a heat balance calculation to account for discrepancies resulting from changes in the control rod pattern and core physics parameters. To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is initially reached and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve may be adjusted to this point, as necessary. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with the predictions.

This process of normalization is generally completed after about 10 percent of the total core burn-up. Thereafter, actual boron concentration is compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1 percent would be unexpected and its occurrence would be thoroughly investigated and evaluated.

3.3.2 Thermal and Hydraulic Tests And Inspections

Fuel thermal and hydraulic tests on models are performed to confirm flow distributions, pressure drops, and departure from nucleate boiling ratios (Reference 2 and Reference 3). Reactor temperature, flow measurements and heat balances are made each reload cycle to confirm the reactor flow rates.

3.3.3 Core Component Tests and Inspections

Quality Assurance and Control is a requirement for all Class I components used in a nuclear reactor. Westinghouse and Framatome ANP fuel has been fabricated under their respective Quality Assurance programs, which have been developed to meet the requirements of 10 CFR 50, Appendix B. Periodic audits have been made by DEK personnel to assure compliance with the Quality Assurance program, including adherence to Quality Control procedures.

Non-fuel Class I core components, such as RCCAs and discrete burnable poison rods, fall under the same type of Quality Assurance and Quality Control requirements. Audits are used to verify compliance with applicable Quality Assurance programs, including adherence to Quality Control procedures.

3.3 References

1. "Reactor Test Program, Kewaunee Power Station," Revision 9, September 2006
2. "HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation Report EMF-92-153(P)(A) and Supplement 1, March 1994
3. Tong, L. S., "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," *Journal of Nuclear Energy*, Vol. 21, pp. 241-248 (1967)